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under the
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VOL. V

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112	Compilation of Nuclear Standards, 9th Edition, 1972, Part I: United States Activities, J. P. Blakely, Oct. 1973	$12.50
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119	A Selected Bibliography on Pressure Vessels for Light-Water-Cooled Power Reactors (LWRs), Fred A. Heddleson, Jan. 1975	$15.00
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VOL. V

(JANUARY—JUNE 1978)

Debbie S. Queener
Engineering Technology Division

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FOREWORD

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VOL. V
(JANUARY–JUNE 1978)

Debbie S. Queener
Engineering Technology Division

ABSTRACT

Lists of documents exchanged during the first half of 1978 under agreements between the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research and the governments of France, Federal Republic of Germany, Japan, and the United Kingdom are presented. These agreements cover safety research on high-temperature gas-cooled reactors, light-water reactors, and fast reactors. During this period, the NRC received 21 reports from France, 87 from the Federal Republic of Germany, 47 from Japan, and 70 from the United Kingdom. In return, the NRC sent 125 U.S. light-water reactor safety research reports to each of these four countries and 22 fast reactor safety research reports to all except France.

INTRODUCTION

This report lists the documents exchanged during the first half of 1978 under agreements between the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research and the governments of France, Federal Republic of Germany, Japan, and the United Kindgom. This is the fifth report in this series. The first had a similar title, but covered the period November 1974 through December 1975 and was published in May 1976 as TID-3362. The second report covered 1976 and was published in April 1977 as ORNL/NUREG/NSIC-134. The third report covered the first half of 1977 and was published in October 1977 as ORNL/NUREG/NSIC-142. The fourth report covered the second half of 1977 and was published in June 1978 as ORNL/NUREG/NSIC-146. The latter half of 1977 witnessed the extension of the light-water reactor (LWR) exchange agreements to include the United Kingdom and the implementation of exchange agreements on both high-temperature gas-cooled reactor (HTGR) and fast reactor safety research. The HTGR safety research exchange agreement is between the U.S. and Japan. The fast reactor safety research exchange agreements are between the U.S. and the Federal Republic of Germany, Japan, and the United Kingdom, respectively.

The total number of reports received by the NRC from January through June 1978 from France, the Federal Republic of Germany, Japan, and the United Kingdom, as well as the reports sent by the NRC to each of these four countries during this period are listed in Table 1 for the LWR, HTGR, and fast reactor exchanges. Also tabulated are the documents exchanged in 1974 and 1975 as listed in TID-3362, in 1976 as reported in ORNL/NUREG/NSIC-134, in the first half of 1977 as reported in ORNL/NUREG/NSIC-142, and in the second half of 1977 as reported in ORNL/NUREG/NSIC-146. The number of proprietary documents received are listed in parentheses.

For convenience in processing, each of the foreign reports received under the exchange agreements is assigned a unique number identifying it as part of the exchange. The documents from France, the Federal Republic of Germany, Japan, and the United Kingdom are listed in the computerized bibliography first by country, then alphabetically by installation, and finally chronologically by report date; the LWR and fast reactor reports are listed separately. Any reports concerning general safety research and those of no specific reactor type that were received under the LWR exchange are listed in the LWR category. Additional bibliographic information, including the number assigned by the issuing installation, abstract, and keywords, is also presented.

The U.S. reports transmitted to foreign exchange agreement countries are listed alphabetically by organization and then alphanumerically by report number. The LWR and fast reactor reports are listed separately.

All reports, both foreign and domestic, with the exception of proprietary documents, are available from the National Technical Information Service, U.S. Department of Commerce, Springfield, VA 22161, in either microfiche or hard copy. (In most cases, however, the hard copies of foreign reports will be reproduced from microfiche.)

Table 1

Reports received by NRC[a]

	France			F.R. Germany			Japan			United Kingdom		
	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor
January–June 1978	21(3)	0	0	87	0	0	45(22)	0	2	62(55)	0	8
July–December 1977	1	0	0	40	0	0	11	0	0	0	0	0
January–June 1977	41(12)	0	0	29(1)	0	0	24(1)	0	0	0	0	0
1976	25(6)	0	0	75(9)	0	0	38(5)	0	0	0	0	0
1974–1975	10	0	0	6	0	0	24	0	0	0	0	0

[a] The number of proprietary documents received are listed in parentheses.

Reports sent by NRC

	France			F.R. Germany			Japan			United Kingdom		
	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor
January–June 1978	125[a]	0	0	125[a]	0	22	125[a]	0	22	125[a]	0	22
July–December 1977	112	0	0	112	0	0	112	37	0	23	0	0
January–June 1977	107	0	0	107	0	0	107	0	0	107	0	0
1976	119	0	0	154	0	0	155	0	0	0	0	0
1974–1975	100	0	0	181	0	0	115	0	0	0	0	0

[a] Includes 4 reports which include LWR, HTGR, and fast reactor safety information.

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NRC LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS TRANSMITTED BY THE NRC TO FOREIGN EXCHANGE AGREEMENT COUNTRIES

The following is a cumulative listing of the U.S. reports transmitted by the NRC to France, the Federal Republic of Germany, Japan, and the United Kingdom during the period January—June 1978.

Argonne National Laboratory

ANL-77-59. *Light-Water-Reactor Safety Research Program: Quarterly Progress Report, April—June 1977.* 1977. Transmitted April 1978.

Battelle Columbus Laboratory

BMI-NUREG-1981. *Fission Product Transport Analysis, Quarterly Progress Report for April through June 1977.* September 30, 1977. Transmitted February 1978.

BMI-NUREG-1985. *Evaluating Strength and Ductility of Irradiated Zircaloy, Quarterly Progress Report July—September 1977.* October 1977. Transmitted March 1978.

BMI-NUREG-1987. *Steam-Water Mixing and System Hydrodynamics Program, Task 4, Quarterly Progress Report July 1, 1977—September 30, 1977.* December 1977. Transmitted March 1978.

BMI-NUREG-1988. *Fission Product Transport Analysis, Quarterly Progress Report for July through September 1977.* November 30, 1977. Transmitted April 1978.

BMI-NUREG-1990. *Analysis of Fission Product Transport Under Terminated LOCA Conditions.* December 30, 1977. Transmitted June 1978.

NUREG-0376. *Residual Stresses at Girth-Butt Welds in Pipes and Pressure Vessels, Final Report April 1, 1976—June 30, 1977.* November 1977. Transmitted February 1978.

NUREG/CR-0034. *Steam-Water Mixing and System Hydrodynamics Program, Task 4, Quarterly Progress Report October 1, 1977—December 31, 1977.* March 1978. Transmitted July 1978.

NUREG/CR-0037. *Residual Stresses at Weld Repairs in Pressure Vessels, Quarterly Progress Report September 1—December 1, 1977.* March 1978. Transmitted July 1978.

Battelle Pacific Northwest Laboratories

BNWL-2091. *A Method for Determining the Uncertainty of Gap Conductance Deduced from Measured Fuel Centerline Temperatures.* February 1977. Transmitted April 1978.

BNWL-2285. *Effects of Fill Gas Composition and Pellet Eccentricity.* July 1977. Transmitted April 1978.

NUREG-0250-2. *Program to Develop Acoustic Emission-Flaw Relationship for Inservice Monitoring of Nuclear Pressure Vessels, Progress Report for February 1–July 1, 1977.* November 1977. Transmitted February 1978.

NUREG-0278, Vol. 1. *Technology, Safety, and Costs of Decommissioning Reference Nuclear Fuel Reprocessing Plant.* October 1977. Transmitted February 1978.

NUREG-0278, Vol. 2. *Technology, Safety, and Costs of Decomissioning Reference Nuclear Fuel Reprocessing Plant.* October 1977. Transmitted February 1978.

NUREG-0359. *Steam Generator Tube Integrity Program, Quarterly Report January 1–March 31, 1977.* October 1977. Transmitted February 1978.

NUREG/CR-0008. *Thermal Performance Measurements on Ultimate Heat Sink - Cooling Ponds.* February 1978. Transmitted June 1978.

Brookhaven National Laboratory

BNL-NUREG-50676. *Concepts for Benchmark Problem Development for Fracture Mechanics Application in Safety Evaluation of Nuclear Piping in Subcreep Service.* June 1977. Transmitted March 1978.

BNL-NUREG-50687. *The Propagation of Phase-Change Fronts in Moving Fluids.* August 1977. Transmitted June 1978.

BNL-NUREG-50747.* *Reactor Safety Research Programs, Quarterly Progress Report July 1–September 30, 1977.* December 1977. Transmitted June 1978.

Creare, Inc.

CREARE TN-271. *Preliminary Analysis of Condensation-Induced Transients.* December 1977. Transmitted May 1978.

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*Report contains LWR, HTGR, and fast reactor information.

General Electric Company

GEAP-20244. *Low Cycle Fatigue Evaluation of Primary Piping Materials in a BWR Environment.* September 1977. Transmitted April 1978.

GEAP-21304-4. *BWR Blowdown/Emergency Core Cooling, Fourth Quarterly Progress Report October 1—December 31, 1976.* January 1977. Transmitted March 1978.

GEAP-21656. *Blowdown Flow in the BWR BDHT Test Apparatus.* October 1977. Transmitted June 1978.

GEAP-NUREG-10207-40. *Reactor Primary Coolant System Pipe Rupture Study Progress Report No. 40, April—June 1977.* August 1977. Transmitted June 1978.

GEAP-NUREG-21304-6. *BWR Blowdown/Emergency Core Cooling, Sixth Quarterly Progress Report April 1—June 30, 1977.* July 1977. Transmitted March 1978.

IIT Research Institute

NUREG/CR-0075. *Accidental Vapor Phase Explosions on Transportation Routes Near Nuclear Power Plants, Final Report January—April 1977.* May 1978. Transmitted August 1978.

Idaho National Engineering Laboratory

TREE-NUREG-1044. *Irradiation Effects Test Series, Scoping Test 2 Test Results Report.* September 1977. Transmitted March 1978.

TREE-NUREG-1066. *Irradiation Effects Test Series, Scoping Test 1 Test Results Report.* September 1977. Transmitted February 1978.

TREE-NUREG-1088. *Experiment Data Report, Gap Conductance Test Series, Test GC 1-3 Postirradiation Examination.* September 1977. Transmitted March 1978.

TREE-NUREG-1106. *Irradiation Effects Test Series, Test IE-3 Test Results Report.* October 1977. Transmitted March 1978.

TREE-NUREG-1109. *Principles of Operation and Data Reduction Techniques for the LOFT Drag Disc Turbine Transducer.* September 1977. Transmitted February 1978.

TREE-NUREG-1111. *LOFT Three-Beam Densitometer Data Interpretation.* October 1977. Transmitted April 1978.

TREE-NUREG-1117. *Investigation of Alternate ECC Injection Concepts in the Semiscale MOD-1 System.* September 1977. Transmitted March 1978.

TREE-NUREG-1124. *Experiment Data Report for Semiscale MOD-1 Test S-06-4 (LOFT Counterpart Test).* December 1977. Transmitted May 1978.

TREE-NUREG-1133. *SCORE-EVET: A Computer Code for the Multidimensional Transient Thermal-Hydraulic Analysis of Nuclear Fuel Rod Arrays.* February 1978. Transmitted June 1978.

TREE-NUREG-1137. *Posttest RELAP4 Analysis of LOFT Experiment L1-3A.* October 1977. Transmitted April 1978.

TREE-NUREG-1148. *Experiment Data Report for Semiscale MOD-1 Test S-28-1 (Steam Generator Tube Rupture Test).* October 1977. Transmitted February 1978.

TREE-NUREG-1149. *Experiment Data Report for Semiscale MOD-1 Test S-28-2 (Steam Generator Tube Rupture Test).* October 1977. Transmitted February 1978.

TREE-NUREG-1150. *Experiment Data Report for Semiscale MOD-1 Test S-28-3 (Steam Generator Tube Rupture Test).* October 1977. Transmitted February 1978.

TREE-NUREG-1151. *Experiment Data Report for Semiscale MOD-1 Test S-28-4 (Steam Generator Tube Rupture Test).* October 1977. Transmitted February 1978.

TREE-NUREG-1152. *Experiment Data Report for Semiscale MOD-1 Test S-28-5 (Steam Generator Tube Rupture Test).* November 1977. Transmitted March 1978.

TREE-NUREG-1153. *Experiment Data Report for Semiscale MOD-1 Test S-28-6 (Steam Generator Tube Rupture Test).* November 1977. Transmitted March 1978.

TREE-NUREG-1154. *Experiment Data Report for Semiscale MOD-1 Test S-28-7, S-28-9, and S-28-12 (Steam Generator Tube Rupture Tests).* February 1978. Transmitted June 1978.

TREE-NUREG-1158. *Axial Gas Flow in Irradiated PWR Fuel Rods.* September 1977. Transmitted February 1978.

TREE-NUREG-1165. *Effects of Intact Loop Hydraulic Resistance of PWR LOCA Behavior in Scaled Experiment Facilities.* September 1977. Transmitted April 1978.

TREE-NUREG-1173. *Experiment Data Report IFA-226 Postirradiation Examination.* September 1977. Transmitted June 1978.

TREE-NUREG-1177. *Postirradiation Examination Results for the Irradiation Effects Test Series IE-ST-2 Rod IE-002.* December 1977. Transmitted May 1978.

TREE-NUREG-1183. *Posttest RELAP4 Analysis of LOFT Experiment L1-4.* November 1977. Transmitted April 1978.

TREE-NUREG-1187. *Power-Cooling-Mismatch Test Series, Test PCM-3, Postirradiation Examination.* December 1977. Transmitted June 1978.

TREE-NUREG-1188. *Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, July—September 1977.* December 1977. Transmitted May 1978.

TREE-NUREG-1191. *LOFT Broken Loop Mass Flow Measurements Uncertainty Analysis.* November 1977. Transmitted March 1978.

TREE-NUREG-1192. *An Assessment of Oxygen Diffusion During UO_2-Zircaloy Interaction.* January 1978. Transmitted June 1978.

TREE-NUREG-1196. *Response of Unirradiated and Irradiated PWR Fuel Rods Tested Under Power-Cooling-Mismatch Conditions.* January 1978. Transmitted May 1978.

TREE-NUREG-1199. *Postirradiation Examination Results for the Irradiation Effects Test IE-1.* February 1978. Transmitted August 1978.

TREE-NUREG-1201. *Postirradiation Examination Results for the Irradiation Effects Test IE-5.* March 1978. Transmitted August 1978.

TREE-NUREG-1202. *CONTEMPT4/MOD2: A Multicompartment Containment System Analysis Report.* February 1978. Transmitted June 1978.

TREE-NUREG-1209. *Experiment Prediction for LOFT Nonnuclear Experiment L1-5.* March 1978. Transmitted August 1978.

Los Alamos Scientific Laboratory

LA-6994. *K-TIF: A Two-Fluid Computer Program for Downcomer Flow Dynamics.* January 1978. Transmitted June 1978.

LA-7039-PR.* *Nuclear Reactor Safety Quarterly Progress Report, July 1—September 30, 1977.* January 1978. Transmitted June 1978.

LA-7089-MS. *Liner-Concrete Heat Transfer Study for Nuclear Power Plant Containments.* January 1978. Transmitted May 1978.

LA-NUREG-6713. *Decay Heat from Products of ^{235}U Thermal Fission by Fast-Response Boil-Off Calorimetry.* September 1977. Transmitted March 1978.

LA-NUREG-6934-PR. *Nuclear Reactor Safety Quarterly Progress Report, April 1—June 30, 1977.* August 1977. Transmitted March 1978.

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*Report contains LWR, HTGR, and fast reactor information.

LA-NUREG-6954-MS. *POST: A Postprocessor Computer Code for Producing Three-Dimensional Movies of Two-Phase Flow in a Reactor Vessel.* September 1977. Transmitted March 1978.

Massachusetts Institute of Technology

NUREG/CR-0003. *Small Scale Modeling of Hydrodynamic Forces in Pressure Suppression Systems, Final Report.* March 1978. Transmitted June 1978.

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NRL/NUREG Report 8165. *Significance of Warm Prestress to Crack Initiation During Thermal Shock.* September 29, 1977. Transmitted March 1978.

NRL Report 8136. *Evaluation of Commercial Production A533-B Steel Plates and Weld Deposits with Extra-Low Copper Content for Radiation Resistance.* October 1977. Transmitted March 1978.

Nebraska Geological Survey

NUREG/CR-0053. *Regional Tectonics and Seismicity of Eastern Nebraska, Annual Report June 1, 1976–June 1, 1977.* April 1978. Transmitted August 1978.

Nuclear Regulatory Commission

NUREG-0193. *FRANTIC — A Computer Code for Time Dependent Unavailability Analysis.* October 1977. Transmitted March 1978.

NUREG-0258. *The OCTAVIA Computer Code: PWR Reactor Pressure Vessel Failure Probabilities Due to Operationally Caused Pressure Transients.* March 1978. Transmitted June 1978.

NUREG-0375. *Two-Phase Flow Instrumentation Review Group Meeting, Proceedings of Meeting January 13–14, 1977.* March 1978. Transmitted June 1978.

NUREG-0418. *Fission Gas Release From Fuel at High Burnup.* March 1978. Transmitted June 1978.

NUREG-0438. *Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants.* April 12, 1978. Transmitted July 1978.

NRC, Electric Power Research Institute, and Westinghouse Electric Corporation

NRC/EPRI/Westinghouse Report No. 1. *PWR FLECHT Separate Effects and System Effects Test (SEASET) Program Plan.* December 1978. Transmitted July 1978.

Oak Ridge National Laboratory

NUREG/CR-0091. *Fission Product Source Terms for the LWR Loss-of-Coolant Accident: Summary Report.* June 1978. Transmitted August 1978.

NUREG/CR-0094. *PWR Blowdown Heat Transfer Separate-Effects Program — Thermal-Hydraulic Test Facility Experimental Data Report for Test 166S.* May 1978. Transmitted August 1978.

ORNL/NUREG-4. *Stresses in Reinforced Nozzle-Cylinder Attachments Under Internal Pressure Loading Analyzed by the Finite-Element Method — A Parameter Study.* October 1977. Transmitted March 1978.

ORNL/NUREG-9. *Test of 6-in.-thick Pressure Vessels, Series 3: Intermediate Test Vessel V-7A Under Sustained Loading.* February 1978. Transmitted June 1978.

ORNL/NUREG-14. *Fission-Product Energy Release for Times Following Thermal-Neutron Fission of* ^{235}U *Between 2 and 14000 Seconds.* October 1977. Transmitted March 1978.

ORNL/NUREG-15. *Significance of Reheat Cracks to the Integrity of Pressure Vessels for Light-Water Reactors.* July 1977. Transmitted March 1978.

ORNL/NUREG-18/V1. *Stress Analysis of Cylindrical Pressure Vessels with Closely Spaced Nozzles by the Finite-Element Method. Volume 1. Stress Analysis of Vessels with Two Closely Spaced Nozzles Under Internal Pressure.* November 1977. Transmitted March 1978.

ORNL/NUREG-19. *PWR Blowdown Heat Transfer Separate-Effects Program Data Evaluation Report — System Response for Thermal-Hydraulic Test Facility Test Series 100.* November 1977. Transmitted March 1978.

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ORNL/NUREG-24. *Experimental Study of Plastic Responses of Pipe Elbows.* February 1978. Transmitted June 1978.

ORNL/NUREG-31. *Zirconium Metal-Water Oxidation Kinetics, V. Oxidation of Zircaloy in High Pressure Steam.* December 1977. Transmitted March 1978.

ORNL/NUREG/CSD-1. *Applications of Kalman Filtering to Nuclear Material Control.* October 1977. Transmitted March 1978.

ORNL/NUREG/NSIC-142. *Reports Distributed Under the NRC Light-Water Reactor Safety Research Foreign Technical Exchange Program, Vol. III (January–June 1977).* October 1977. Transmitted March 1978.

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ORNL/NUREG/TM-119. *Quarterly Progress Report on the Advanced Two-Phase Instrumentation Program for January–March 1977.* July 1977. Transmitted March 1978.

ORNL/NUREG/TM-121. *Fission Product Beta and Gamma Energy Release, Quarterly Progress Report for January–March 1977.* July 1977. Transmitted March 1978.

ORNL/NUREG/TM-122. *Quarterly Progress Report on Fission Product Behavior in LWRs for the Period January–March 1977.* June 1977. Transmitted March 1978.

ORNL/NUREG/TM-125. *Quarterly Progress Report on the Creepdown and Collapse of Zircaloy Fuel Cladding for July 1976–March 1977.* July 1977. Transmitted March 1978.

ORNL/NUREG/TM-132. *Quarterly Progress Report on the Zirconium Metal-Water Oxidation Kinetics Program Sponsored by the NRC Division of Reactor Safety Research for April–June 1977.* August 1977. Transmitted March 1978.

ORNL/NUREG/TM-133. *Noise Diagnostics for Safety Assessment, Quarterly Progress Report for April–June 1977.* September 1977. Transmitted March 1978.

ORNL/NUREG/TM-134. *Quarterly Progress Report on Blowdown Heat Transfer Separate-Effects Program for April–June 1977.* November 1977. Transmitted March 1978.

ORNL/NUREG/TM-135. *Multirod Burst Test Program, Quarterly Progress Report for April–June 1977.* December 1977. Transmitted March 1978.

ORNL/NUREG/TM-140. *Advanced Two-Phase Instrumentation Program, Quarterly Progress Report for April–June 1977.* December 1977. Transmitted March 1978.

ORNL/NUREG/TM-145. *Analysis of Design Practices for Snubbers (Report 3).* January 1978. Transmitted May 1978.

ORNL/NUREG/TM-149. *Quarterly Progress Report on Blowdown Heat Transfer Separate-Effects Program for July–September 1977.* March 1978. Transmitted August 1978.

ORNL/NUREG/TM-156. *Fission Product Beta and Gamma Energy Release, Quarterly Progress Report for April—June 1977.* November 1977. Transmitted March 1978.

ORNL/NUREG/TM-159. *Quarterly Progress Report on the Zirconium Metal-Water Oxidation Kinetics Program Sponsored by the NRC Division of Reactor Safety Research for July—September 1977.* November 1977. Transmitted March 1978.

ORNL/NUREG/TM-161. *Noise Diagnostics for Safety Assessment, Quarterly Progress Report for July—September 1977.* November 1977. Transmitted June 1978.

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ORNL/NUREG/TM-170. *Quarterly Progress Report on Fission Product Behavior in LWRs for the Period July—September 1977.* January 1978. Transmitted May 1978.

ORNL/NUREG/TM-176. *Noise Diagnostics for Safety Assessment, Quarterly Progress Report for October—December 1977.* February 1978. Transmitted May 1978.

ORNL/NUREG/TM-180. *Critical Heat Flux Experimentation in an Annular Test Section.* March 1978. Transmitted June 1978.

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Oklahoma Geological Survey

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Oregon State University

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Rensselaer Polytechnic Institute

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Sandia Laboratories

NUREG-76-6521. *Radiation Signature Following the Hypothesized LOCA.* September 1977. Transmitted March 1978.

NUREG-0183-3. *Light Water Reactor Safety Research Program, Quarterly Report for January–March 1977.* October 1977. Transmitted March 1978.

SAND77-0922.* *Radiant Heat Evaluation of Concrete — A Study of the Erosion of Concrete Due to Surface Heating.* January 1978. Transmitted June 1978.

SAND77-1467.* *Response of Concrete Exposed to a High Heat Flux on One Surface.* November 1977. Transmitted June 1978.

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NUREG-0264-3. *Suppression Pool Dynamics, Annual Report July 1, 1976–June 30, 1977.* February 1978. Transmitted June 1978.

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NUREG/CR-0039. *Studies of Dynamic Loads in Pressure Suppression Containment, Quarterly Report July 1–September 30, 1977.* March 1978. Transmitted June 1978.

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NUREG-0342. *Photoelastic Studies of Crack Propagation and Crack Arrest.* October 1977. Transmitted March 1978.

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*Report contains LWR, HTGR, and fast reactor information.

University of Virginia

NUREG-0391. *Primary Aerosol Particle-Size Distribution from Homogeneous Nucleation Condensation and Particle Growth.* December 1977. Transmitted March 1978.

Westinghouse Electric Corporation

WCAP-9183. *PWR FLECHT Skewed Profile Low Flooding Rate Test Series Evaluation Report.* November 1977. Transmitted July 1978.

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The following is a cumulative listing of the U.S. reports transmitted by the NRC to the Federal Republic of Germany, Japan, and the United Kingdom during the period January—June 1978.

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ANL-77-27. *Physics Calculations for the Clinch River Breeder Reactor.* June 1977. Transmitted May 1978.

ANL-77-69. *Physics of Reactor Safety, Quarterly Report April—June 1977.* September 1977. Transmitted May 1978.

ANL-77-92. *Physics of Reactor Safety, Quarterly Report July—September 1977.* December 1977. Transmitted May 1978.

NUREG-0333. *Bubble Dynamics in a Superheated Liquid.* September 1977. Transmitted May 1978.

NUREG-0355. *Numerical Results Obtained from the Three Dimensional Transient Single Phase Version of the COMMIX Computer Code.* October 1977. Transmitted May 1978.

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BMI-NUREG-1983. *Aerosol Measurements and Modeling for Fast Reactor Safety: Quarterly Progress Report for April 1 through June 30, 1977.* October 30, 1978. Transmitted March 1978.

BMI-NUREG-1989. *Aerosol Measurements and Modeling for Fast Reactor Safety: Annual Report for FY1977.* December 1977. Transmitted May 1978.

BMI-NUREG-1991. *HAARM-3 Users Manual.* January 5, 1978. Transmitted July 1978.

Battelle Pacific Northwest Laboratories

NUREG-0387. *A Corrosion Critique of the 2 1/4 Cr—1 Mo Steel for LMFBR Steam Generation System Applications.* November 1977. Transmitted February 1978.

Brookhaven National Laboratory

BNL-NUREG-50688. *A Pipe-Break Model for LMFBR Safety Analysis.* August 1977. Transmitted May 1978.

BNL-NUREG-50722. *Heat Transfer and Fluid Dynamics Characteristics of Internally Heated Boiling Pools.* September 1977. Transmitted May 1978.

Los Alamos Scientific Laboratory

LA-7133-MS. *A Feasibility Study for a Postaccident Heat Removal Facility.* January 1978. Transmitted July 1978.

Mound Laboratory

MLM-MU-77-70-0004. *Annual Report Aerosol Characterization from a Simulated HCDA.* October 27, 1977. Transmitted March 1978.

Oak Ridge National Laboratory

ORNL/NUREG/TM-124. *Source Term Assessment Concepts for LMFBRS: Aerosol Release and Transport (ART) Analytical Program.* November 1977. Transmitted May 1978.

ORNL/NUREG/TM-142. *Breeder Reactor Aerosol Release and Transport Program Quarterly Progress Report for April—June 1977.* December 1977. Transmitted March 1978.

ORNL/NUREG/TM-165. *Inventory and Relative Biological Hazard of LMFBR Core Materials.* January 1978. Transmitted July 1978.

Sandia Laboratories

NUREG-0120-4. *Annular Core Pulse Reactor Upgrade, Quarterly Report July—September 1977.* December 1977. Transmitted May 1978.

SAND77-1133. *Annular Core Pulse Reactor Upgrade, Quarterly Report April—June 1977.* September 1977. Transmitted May 1978.

SAND77-1134. *Advanced Reactor Safety Research Program, Quarterly Report April—June 1977.* November 1977. Transmitted April 1978.

University of Arizona

NUREG/CR-0016. *A Parameter Sensitivity Study in the Dynamic Simulation of a Sodium Cooled Breeder Reactor Power Plant.* February 1978. Transmitted July 1978.

University of Missouri-Columbia

NUREG/CR-0015. *Analysis of Aerosol Particles Undergoing Gravitational Agglomeration, Annual Progress Report December 15, 1976–September 30, 1977.* January 1978. Transmitted July 1978.

University of Virginia

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NUREG/CR-0002. *Bubble Behavior in LMFBR Core-Disruptive Accidents, Annual Report July 1, 1976–September 30, 1977.* February 1978. Transmitted July 1978.

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ORGANIZATION OF BIBLIOGRAPHY

The bibliography which follows contains all the safety research reports received by NRC under the foreign exchange agreement program from January through June 1978. The reports received from France, Japan, and the United Kingdom are listed chronologically by report date, and the German reports are listed alphabetically by installation name and then chronologically by report date under each installation. The bibliography is sorted into the following categories:

1. French light-water reactor safety research reports received by NRC.
2. German (FRG) light-water reactor safety research reports received by NRC.
3. Japanese light-water reactor safety research reports received by NRC.
4. Japanese fast reactor safety research reports received by NRC.
5. U.K. light-water reactor safety research reports received by NRC.
6. U.K. fast reactor safety research reports received by NRC.

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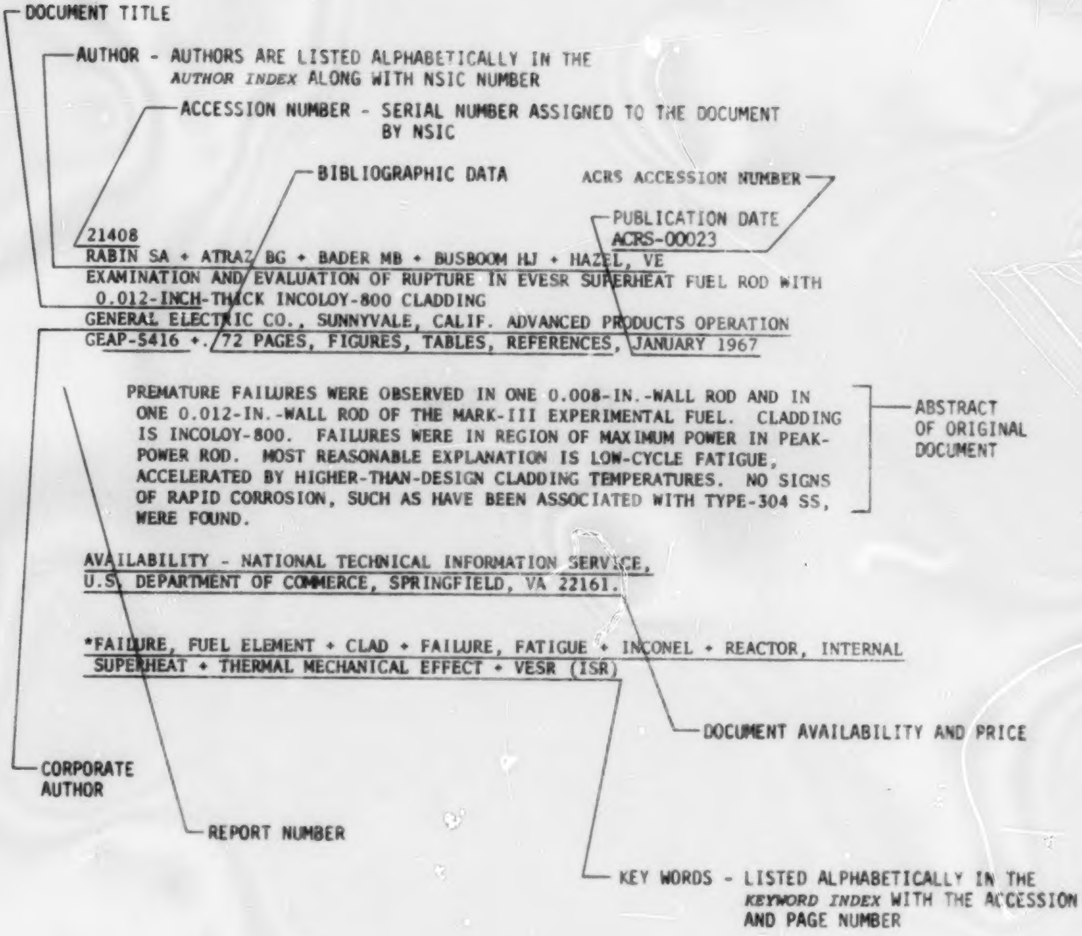

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AUTHOR - AUTHORS ARE LISTED ALPHABETICALLY IN THE *AUTHOR INDEX* ALONG WITH NSIC NUMBER

ACCESSION NUMBER - SERIAL NUMBER ASSIGNED TO THE DOCUMENT BY NSIC

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ACRS ACCESSION NUMBER

PUBLICATION DATE

21408
ACRS-00023
RABIN SA + ATRAZ BG + BADER MB + BUSBOOM HJ + HAZEL VE
EXAMINATION AND EVALUATION OF RUPTURE IN EVESR SUPERHEAT FUEL ROD WITH 0.012-INCH-THICK INCOLOY-800 CLADDING
GENERAL ELECTRIC CO., SUNNYVALE, CALIF. ADVANCED PRODUCTS OPERATION
GEAP-5416 +. 72 PAGES, FIGURES, TABLES, REFERENCES, JANUARY 1967

PREMATURE FAILURES WERE OBSERVED IN ONE 0.008-IN.-WALL ROD AND IN ONE 0.012-IN.-WALL ROD OF THE MARK-III EXPERIMENTAL FUEL. CLADDING IS INCOLOY-800. FAILURES WERE IN REGION OF MAXIMUM POWER IN PEAK-POWER ROD. MOST REASONABLE EXPLANATION IS LOW-CYCLE FATIGUE, ACCELERATED BY HIGHER-THAN-DESIGN CLADDING TEMPERATURES. NO SIGNS OF RAPID CORROSION, SUCH AS HAVE BEEN ASSOCIATED WITH TYPE-304 SS, WERE FOUND.

ABSTRACT OF ORIGINAL DOCUMENT

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA 22161.

*FAILURE, FUEL ELEMENT + CLAD + FAILURE, FATIGUE + INCONEL + REACTOR, INTERNAL SUPERHEAT + THERMAL MECHANICAL EFFECT + VESR (ISR)

DOCUMENT AVAILABILITY AND PRICE

CORPORATE AUTHOR

REPORT NUMBER

KEY WORDS - LISTED ALPHABETICALLY IN THE *KEYWORD INDEX* WITH THE ACCESSION AND PAGE NUMBER

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BIBLIOGRAPHY

1. FRENCH LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM FRANCE DURING THE FIRST HALF OF 1978 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

137797
DOURY A + GERARD R + PICOL M
GRAPHS FOR THE DIRECT EVALUATION OF ATMOSPHERIC TRANSFERS OF GASEOUS EFFLUENTS (IN FRENCH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
DSN-84(REV.1) + FRRSR-78 +. 96 PPS, 85 FIGS, 4 REFS, MARCH 1977

GRAPHS ARE PRESENTED WHICH OFFER AN ECONOMICAL AND CONVENIENT MEANS FOR PREDICTING RAPIDLY AND QUANTITATIVELY THE POLLUTIONS TRANSPORTED AND DIFFUSED BY THE ATMOSPHERE. THE RESULTS ARE VALID FOR THE GROUND LEVEL AND DIFFERENT POINTSOURCE HEIGHTS, ALONG THE AVERAGE WIND DIRECTION AS WELL AS CROSSWIND.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ATMOSPHERIC DIFFUSION + ATMOSPHERIC POLLUTION + EFFLUENT + DIFFUSION + TRANSPORT + FOREIGN EXCHANGE

137612
RINGOT C
SAFETY STUDIES OF THERMONUCLEAR REACTORS IN FRANCE (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 143 + FRRSR-79 +. APPROX. 50 PPS, FIGS, REFS, PAPER PRESENTED AT IAEA CONSULTING GROUP MEETING: FEB. 14-18, 1977

ONLY A FEW STUDIES ARE BEING PERFORMED AT THE PRESENT TIME, DEALING WITH THE DEVELOPMENT RATHER THAN WITH THE SAFETY OF REACTORS. SAFETY PROGRAMS HAVE BEEN STARTED IN FRANCE MAINLY TO COPE WITH THE PROBLEM OF FISSION PRODUCT RELEASE IN THE PRIMARY CIRCUIT OF HIGH TEMPERATURE REACTORS. THE FRENCH ELECTRO-NUCLEAR PROGRAM IS BASED PRINCIPALLY ON THE CONSTRUCTION OF PRESSURIZED WATER REACTORS, UNDER LICENCE FROM WESTINGHOUSE, AND ON SODIUM-COOLED FAST REACTORS. THE PROGRAM OF PLANT ORDERS HAS BEEN CONSIDERABLY STEPPED UP SINCE 1974 FOLLOWING THE CRISIS DUE TO THE OIL-PRICE INCREASE, WITH THE AIM OF HAVING NUCLEAR POWER MEET ABOUT 25% OF ELECTRICITY DEMAND BY 1985. THE SHARE OF ELECTRICITY PRODUCTION SHOULDERED BY NUCLEAR ENERGY PRESENTLY REACHES 2.5%, WITH 8 GAS-COOLED REACTORS, 1 PWR (270 MWE) AND THE FAST NEUTRON REACTOR NAMED "PHENIX" (250 MWE).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REVIEW + R AND D PROGRAM + *FRANCE + REACTOR, POWER + *SAFETY PROGRAM + FOREIGN EXCHANGE

106295
SAGLIO R + PROT AC
"EFFECTIVE ULTRASONIC BEAM" CHARACTERISTICS AS THE BASIS FOR OBJECTIVE NOT SPECIFICATIONS
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 149E + FRRSR-80 +. 17 PPS, 5 FIGS, 7 REFS, SEPT. 27, 1976

SEVERAL DIFFICULTIES ARE ASSOCIATED WITH THE USE OF ULTRASONIC NONDESTRUCTIVE TESTING METHODS. BUT DUE TO THE SENSITIVITY OF THESE METHODS TO DETECT CRACK LIKE DEFECTS, THEY ARE OF COMMON USE IN MOST OF THE ADVANCED INDUSTRIES. AFTER REVIEWING THE MAIN SOURCES OF DIFFICULTIES, IT IS SHOWN HOW THE USE OF FOCUSSED PROBES HAS LED TO THE CONCEPT OF THE "EFFECTIVE ULTRASONIC BEAM". THIS CONCEPT ALLOWS AN OBJECTIVE DETERMINATION OF THE DEFECT SIZE WITH A GIVEN AND KNOWN ACCURACY. IT SHOWS ALSO THAT THE USE OF FOCUSSED PROBES ENABLE A GOOD LONG TERM REPRODUCIBILITY AND A LOWER SENSITIVITY TO DEFECT ORIENTATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ULTRASONICS + TEST, NONDESTRUCTIVE + *INSERVICE INSPECTION + DESIGN CRITERIA + FOREIGN EXCHANGE

134160
BHANDARI SK + BARRACHIN B
ANALYSIS OF A CRACKED-MEDIUM UNDER FIXED-DEFORMATION LOADING
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 151E + FRRSR-81 +. 9 PPS, FROM 3RD INTERNATIONAL CONFERENCE ON PRESSURE VESSEL TECHNOLOGY; TOKYO, JAPAN, APRIL 18-22, 1977

THE PRESENT INVESTIGATION DEALS WITH THE PROBLEM OF AN ELASTIC CRACKED MEDIUM UNDER IMPOSED DEFORMATION, ELUCIDATED BY CONSIDERING THE CASE OF A PLATE UNDER GENERAL DEFORMATION LOADING. THE VALUES OF THE STRESS-INTENSITY FACTOR K HAVE BEEN EVALUATED FOR VARIOUS GEOMETRICAL AND LOADING CONDITIONS AND HAVE BEEN COMPARED WITH THE EQUIVALENT CASE OF THE PLATE UNDER FIXED FORCE CONDITIONS. THE STRAIN ENERGY DENSITY FACTOR AS PROPOSED BY SIH HAS ALSO BEEN DERIVED FOR UNIAXIAL DEFORMATION AND QUALITATIVELY COMPARED WITH THE FACTOR K.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CRACK + PLATE + DEFORMATION + STRESS ANALYSIS + FOREIGN EXCHANGE

135136
PROT AC + SAGLIO R
SURVEY OF THE FRENCH DEVELOPMENTS IN THE FIELD OF IN-SERVICE INSPECTION

135136 *CONTINUED*
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
DSN 152E + FRRSR-82 +. 47 PPS, FIGS, APRIL 25, 1977

THE FRENCH PROGRAM FOR PWR NUCLEAR PLANTS WAS AT THE ORIGIN OF A LARGE AMOUNT OF R AND D WORK IN THE FIELD OF IN-SERVICE INSPECTION. THE ACTIONS ARE: THE REGULATORY LEVEL - A GOVERNMENT ORDER STATES PROVISIONS TO BE APPLIED TO THE MAIN PRIMARY CIRCUIT OF NUCLEAR WATER STEAM SUPPLY SYSTEMS. THE ADVANTAGES OF THE FRENCH PHILOSOPHY ARE POINTED OUT; THE LEVEL OF THE R AND D WORK - METHODS AND DEVICES WERE STUDIED TO IMPROVE THE RELIABILITY OF NDT TECHNIQUES; THE DESIGN LEVEL - ACTIONS UNDERTAKEN TO FACILITATE OR IMPROVE THE TESTING DURING IN SERVICE INSPECTION ARE PRESENTED; THE FLAW EVALUATION LEVEL - IT IS SHOWN HOW THE PRESENTED METHODS MAY IMPROVE THE EVALUATION TASK; FINALLY THE FIRST RESULTS OF IN-SERVICE INSPECTIONS ARE PRESENTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SURVEY + FRANCE + *INSERVICE INSPECTION + *R AND D PROGRAM + REACTOR, PWR + TEST, NONDESTRUCTIVE + FOREIGN EXCHANGE

137457
PELCE J + NOC B + TROCCON G
STUDIES PERFORMED IN FRANCE ON THE BLOWDOWN ACCIDENT IN LIGHT WATER REACTORS (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN-154 + FRRSR-83 +. 12 PPS, 5 REFS, MAY 1977

REPORT DESCRIBING EFFORTS UNDERWAY IN FRANCE TO IMPROVE ANALYTICAL TECHNIQUES USED IN EVALUATING THE EFFECTS OF A LOSS OF COOLANT ACCIDENT. THE APPROACH IS A COMBINATION OF ANALYTICAL PROGRAMS WITH EXPERIMENTS. INCLUDED IN THE PROGRAMS ARE: CRITICAL FLOWS AT THE BREAK AND BETWEEN HOT CAVES INSIDE THE CONTAINMENT, WALL CONDENSATION, PHASE SEPARATION IN THE EVENT OF SMALL BREAKS, WATER-STEAM MIXING DURING EMERGENCY INJECTION, BEHAVIOR OF FUEL CLADDING, HEAT TRANSFER INSIDE THE CORE DURING BLOWDOWN AND REFLOOD, AND FUEL BEHAVIOR DURING THE ACCIDENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*REACTOR, LWR + FRANCE + *BLOWDOWN + *ACCIDENT, LOSS OF COOLANT + ANALYTICAL MODEL + ANALYTICAL TECHNIQUE + EXPERIMENT + COMPUTER PROGRAM + CORE REFLOODING + FOREIGN EXCHANGE

132813
CARNINO A + COUDERT C + GACHOT B + GREPPO JF
STATUS AND OUTLOOK OF RELIABILITY STUDIES IN THE NUCLEAR FIELD IN FRANCE
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 156 + FRRSR-94 +. 14 PPS, 25 REFS, MAY 2, 1977

IN FRANCE, THE BUILDERS (MAINLY FRAMATOME AND GAAA), THE NUCLEAR POWER PLANT OPERATORS - THAT IS, ELECTRICITE DE FRANCE - AND, OF COURSE, THE SAFETY AUTHORITIES, STARTING FROM COMMON BASIC CONCEPTS HAVE BROUGHT FORTH MANY STUDIES OF PROGRAMMED NUCLEAR SYSTEMS, THUS FORMING TEAMS WHICH, ACTING THROUGH JOINT WORKING GROUPS, PERFORM VARIOUS TASKS IN CLOSE CONTACT WITH EACH OTHER. THE FIELDS OF APPLICATION INCLUDE THE ANALYSIS OF SYSTEMS FOR EVALUATING THESE PERFORMANCES AND LAYING DOWN OPERATIONAL RULES TO FALL BACK UPON WHENEVER SAID SYSTEMS ARE PARTIALLY UNAVAILABLE. THE SYSTEMS ANALYSED IN THIS WAY ARE SAFEGUARD SYSTEMS AND THEIR AUXILIARIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FRANCE + *RELIABILITY ANALYSIS + PROBABILITY + ANALYTICAL TECHNIQUE + FOREIGN EXCHANGE

135152
CARNINO A + DUBAU J
DETERMINATION OF INITIATING EVENTS AND SEQUENCES OF REACTOR ACCIDENTS BY A BARRIER ANALYSIS
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 163E + FRRSR-85 +. 27 PPS, 13 FIG, JUNE 20, 1977

FROM A BARRIER ANALYSIS (FUEL CLADDING, PRIMARY CIRCUIT, CONTAINMENT), A SYSTEMATIC SEARCH FOR THE POTENTIAL INITIATING EVENTS AND THE ACCIDENT SEQUENCES IS CONDUCTED. THE METHOD IS DIVIDED INTO THREE PARTS: THE BARRIER ANALYSIS - FOR EACH BARRIER, AN EVENT TREE IS CONSTRUCTED HAVING FOR TOP EVENT THE BARRIER LOSS; THE SEARCH FOR INITIATING EVENTS OF ACCIDENTS (AN ACCIDENT HAS OCCURRED WHEN ALL THE BARRIERS ARE BROKEN OR FAILED); ACCIDENTS EVENT TREES - FROM AN INITIATING EVENT, THE ACCIDENT SEQUENCE IS DERIVED. TO ILLUSTRATE THE METHOD, THE FAILURE OF ALL THE ELECTRICAL POWER SUPPLIES IN A 900 MW PWR IS CONSIDERED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*RELIABILITY ANALYSIS + *ANALYTICAL TECHNIQUE + ACCIDENT ANALYSIS + FAILURE MODE ANALYSIS + FAULT TREE ANALYSIS + FOREIGN EXCHANGE

134235
GARNIER C + BARRACHIN B + ROCHE R
LOW CYCLE FATIGUE OF AUSTENITIC ALLOYS IN HOT WATER

134235 *CONTINUED*
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
PAPER L 6/7 + FRRSR-86 +. 8 PPS, 7 FIGS, 7 REFS, 1977

TESTS ARE IN PROGRESS ON DEVICES CONNECTED WITH THE "PRIMEAU" LOOP. THIS LOOP PROVIDES CIRCULATING WATER WITH THE REQUIRED CONDITIONS (PRESSURE, TEMPERATURE, CHEMICAL CONTENTS). SOME INDICATIONS ARE GIVEN ON THE MAIN FEATURES OF THIS LOOP, WITH SPECIAL EMPHASIS ON THE TEST SECTIONS. EACH TEST SECTION ALLOWS BATCHES OF FOUR SAMPLES TO BE TESTED AT A TIME. FOR CALIBRATION OF THE RESULTS, SIMILAR TESTS ARE CARRIED OUT IN AIR AT ROOM TEMPERATURE ON ISOLATED TEST SECTIONS. PRELIMINARY RESULTS SHOW THAT LOW CYCLE FATIGUE RESISTANCE IS AFFECTED BY TEST CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STEEL, STAINLESS + CYCLING + FATIGUE + WATER + TEMPERATURE + COMPOSITION + PROPERTY, CHEMICAL + FOREIGN EXCHANGE

134199
GARNIER C + KOWALCZUK G + ROCHE R
LOW CYCLE FATIGUE OF STEELS FOR NUCLEAR PRESSURE VESSELS IN HOT WATER (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
CEA-N-1973 + FRRSR-87 +. 35 PPS, 20 FIGS, JULY 1977

A GENERAL PROGRAM ON LOW CYCLE FATIGUE IN HOT WATER IS IN PROGRESS AT SACLAY TO ASSESS CONSERVATISM OF DESIGN FATIGUE CURVES. TO KEEP CONTROL OF WATER CHEMISTRY, THE TEST RIG IS CONNECTED WITH THE "PRIMEAU" LOOP AND THEREFORE A CONSTANT FLOW OF WATER IS OBTAINED THROUGH THE SPECIMENS. TESTED SAMPLES ARE SMALL STRIPS OF RECTANGULAR CROSS SECTION (2 X 10 X 61 MM). THEY ARE LOADED IN CIRCULAR BENDING. TESTS ARE NOW IN PROGRESS ON AUSTENITIC ALLOYS AND THE AVAILABLE RESULTS ARE GIVEN. TESTS ON A FERRITIC STEEL HAVE BEEN COMPLETED. THERE IS NO NOTICEABLE DIFFERENCE BETWEEN HOT WATER TEST RESULTS AND AIR-ROOM TEMPERATURE RESULTS. IN COMPARISONS MADE WITH ASME CRITERIA, RESULTS ARE NEAR THE ASME BEST FIT CURVE FOR ASTM 392 STEEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CYCLING + STEEL, STAINLESS + COOLANT CHEMISTRY + REACTOR COOLANT + TESTING + FATIGUE + WATER + FOREIGN EXCHANGE

134124
RUZENBERG J + DOLLE L + DARRAS R
CONTROL OF CORROSION PRODUCT MOVEMENTS AND DEPOSITED ACTIVITIES IN REACTOR CIRCUITS
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
PAPER IWC-76-8 + FRRSR-88 +. 19 PPS, PAPER PRESENTED AT 37TH ANNUAL MEETING INTERNATIONAL WATER CONFERENCE; PITTSBURGH, PA., OCT. 26-28, 1976

THE EXPERIMENTAL RESULTS DERIVED FROM CONTROLLED COOLING EXPERIMENTS CONDUCTED ON THE WATER OF AN IN-PILE LOOP CONFIRMED THE COMPLEXITY OF BEHAVIOR OF CORROSION PRODUCTS. THEY GAVE RISE TO THREE BASIC OBSERVATIONS: (1) AS THE PRIMARY WATER IS COOLED, THE ACTIVITIES OF THE MAIN RADIONUCLIDES IN THE WATER INCREASE. (2) WHILE THE CRUD CONTENT INCREASES DURING COOLING, THE MOVEMENTS OF CHEMICAL SPECIES IN THE WATER VARY WIDELY. (3) MOLYBDENUM AND MOLYBDENUM-99 EXHIBIT SURPRISING BEHAVIOR, FREQUENTLY CLEARLY DISTINGUISHED FROM THE OTHER CORROSION PRODUCTS, PARTICULARLY OWING TO THE SPECTACULAR RISE IN SOLUBILIZATION OF THE MOLYBDENUM DEPOSITED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORROSION + OPERATING EXPERIENCE + IN PILE EXPERIMENT + COOLANT CHEMISTRY + CRUD + FOREIGN EXCHANGE + MOLYBDENUM

134152
ROZENBERG J + ROCHE P + KAHAN H
PHYSICO-CHEMICAL OBSERVATIONS ON THE WATER OF THE LOOP IRENE (CYCLE 03) (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
DRA/SAECNI/74-171/JROZ/JG/FN + FRRSR-89 +. 12 PPS, 2 FIGS, 5 REFS, APRIL 1974

THE CYCLE 03 WAS DISTINGUISHED BY GREAT STABILITY OF THE PHYSICO-CHEMICAL PROPERTIES OF THE WATER, AFTER AN INITIAL PERTURBATION FOLLOWING THE SETTING IN PLACE OF NEW RESINS. THE PH OF THE WATER, STABILISED AT 10, KEPT THE SOLUBLE IRON CONCENTRATIONS BELOW 5 PPB, IN AGREEMENT WITH CALCULATIONS. THE MAIN SOURCES OF ACTIVITY WERE: MO-99, CR-51, MN-54, CO-60, CO-58 AND FE-59. A PROGRAMMED COOLING EXPERIMENT, CARRIED OUT DURING THE CYCLE, ALSO REVEALED A SPECIAL PART PLAYED BY THE NUCLIDES AS-76, SB-122 AND SB-124.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORROSION + OPERATING EXPERIENCE + IN PILE EXPERIMENT + COOLANT CHEMISTRY + RESIN + ANALYTICAL TECHNIQUE + MOLYBDENUM + CHROMIUM + MANGANESE + COBALT + IRON + RADIOISOTOPE + FOREIGN EXCHANGE

134151
PAULI O + ROZENBERG J

134151 *CONTINUED*
GENERATION OF HYDROGEN BY RADIOLYTIC DECOMPOSITION OF SPRINKLING SOLUTIONS CAPABLE OF ABSORBING FISSION IODINE. INFLUENCE OF CERTAIN ADDITIVES (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
DRA/SAECNI/75-387/JR/MM + FRRSR-90 +. 18 PPS. 5 FIGS. 5 REFS. JUNE 1975

SPRAYING SOLUTIONS MAY BE USED TO ABSORB FISSION IODINE IN THE EVENT OF ACCIDENTAL LOSS OF COOLANT FLUID, IF THE REACTOR CORE HAS RELEASED FISSION PRODUCTS. HOWEVER THE PRODUCTION OF HYDROGEN, CAUSED BY RADIOLYSIS OF THESE SOLUTIONS, IN THE CONFINEMENT VESSEL OF AN ELECTRONUCLEAR STATION CAN BE A SOURCE OF DANGER. ADDITIVES ARE THEREFORE TESTED WITH A VIEW TO REDUCING THE VOLUME OF HYDROGEN LIBERATED. FIRST OF ALL, THE VARIATIONS IN THE HYDROGEN FORMATION YIELD UNDER GAMMA RADIATIONS FROM A LABORATORY IRRADIATOR ARE STUDIED AS A FUNCTION OF VARIOUS ADDITIVES IN THE LIQUID PHASE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SPRAY + ENGINEERED SAFETY FEATURE + HYDROGEN + IODINE + FOREIGN EXCHANGE + RADIOLYTIC GAS

134155
ROZENBERG J + KAHAN H
PHYSICO-CHEMICAL OBSERVATIONS ON THE WATER OF THE LOOP IRENE (CYCLE 05-1) - CONDITIONING WITH LITHIA AND BORIC ACID (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
DRA/SAECNI/74-491/JR/JS + FRRSR-91 +. 16 PPS. 5 TABS. 9 FIGS. OCT. 1974

AFTER A STABLE RUN WITH LITHIATED WATER, BORIC ACID WAS ADDED FOR THE FIRST TIME TO THE WATER OF THE LOOP IRENE. THE ONLY IMMEDIATE RESULT OF THIS CHANGED CONDITIONING WAS A RELEASE OF IRON FOLLOWED BY REDEPOSITION. FOR THE FIRST TIME THE COBALT CONCENTRATION EXCEEDED THE USUAL DETECTION LIMIT (>10 PPB).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORROSION + IN PILE EXPERIMENT + OPERATING EXPERIENCE + DEPOSITION + COOLANT CHEMISTRY + LITHIUM + BORON + IRON + COBALT + FOREIGN EXCHANGE

134154
KAHAN H + ROZENBERG J
PHYSICO-CHEMICAL OBSERVATIONS ON THE WATER OF THE LOOP IRENE (CYCLE 07) (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
DRA/SAECNI/75.764.511/JR-FMC + FRRSR-92 +. 33 PPS. 10 TABS. 9 FIGS. 6 REFS. DEC. 1975

AFTER A STABLE RUN WITH NATURAL LITHIA-CONDITIONED WATER THE SECOND PART OF THE IRENE 07 CYCLE WAS DEVOTED TO A STUDY ON THE TRITIUM ACTIVITY VARIATION IN THE LOOP. THE BALANCE OF THIS RADIO-ELEMENT WAS TO SERVE FOR THE DETERMINATION OF A FISSION TRITIUM DIFFUSION RATE THROUGH THE ZIRCALOY OF THE FUEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORROSION + IN PILE EXPERIMENT + OPERATING EXPERIENCE + COOLANT CHEMISTRY + DIFFUSION + LITHIUM + TRITIUM + ZIRCALOY + FOREIGN EXCHANGE

134153
KAHAN H + ROZENBERG J
PHYSICO-CHEMICAL OBSERVATIONS ON THE WATER OF THE LOOP IRENE DURING A PROGRAMMED COOLING (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
DRA/SAECNI/74-48/JR/JG + FRRSR-94 +. 24 PPS, 7 FIGS. 2 REFS. JAN. 1974

BY COMPARING THE ACTIVITIES OF THE SOLUBLE AND INSOLUBLE SPECIES, AN INSOLUBILITY CRITERION IN A CONDITIONED WATER AT PH 10 WAS DEFINED. THUS IRON, COBALT, CHROMIUM AND ZIRCONIUM ARE PRESENT MAINLY IN INSOLUBLE FORM, WHEREAS MOLYBDENUM, ARSENIC AND ANTIMONY TEND TO BE SOLUBLE; MANGANESE FALLS IN BETWEEN. FINALLY THE INSENSITIVITY OF IRON TO THE THERMAL CYCLE PROVES THE GREAT CLEANLINESS OF THE WATER IN THE LOOP IRENE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORROSION + IN PILE EXPERIMENT + COOLANT CHEMISTRY + RADIOISOTOPE + IRON + COBALT + CHROMIUM + ZIRCONIUM + MOLYBDENUM + ARSENIC + ANTIMONY + FOREIGN EXCHANGE

135410
FRANK R + MAZARS J + RICOUE R
DETERMINATION OF MASS FLOW RATE AND QUALITY USING A TURBINE METER AND A VENTURI
CEA CENTRE D'ETUDES NUCLEAIRES, GRENOBLE, FRANCE
C208/77 + FRRSR-96 +. 5 PPS. 3 FIGS. 1977

IT IS NECESSARY, IN BLOWDOWN EXPERIMENTS, TO MEASURE ACCURATELY THE MASS FLOW RATE, THE QUALITY AND THE VOID FRACTION IN TWO PHASE FLOW, AT SOME POINTS OF THE INSTALLATION. A METHOD OF

135410 *CONTINUED*

DETERMINING THE MASS FLOW RATE AND THE QUALITY IS PROPOSED USING A VENTURI AND A TURBINE METER. THESE TWO INSTRUMENTS ARE STUDIED AND CALIBRATED IN STEADY STATE TWO PHASE FLOW. A MODEL OF INTERPRETATION OF EACH OF THESE DEVICES IS PROPOSED AND THE PROCEDURE USED TO DETERMINE THE MASS FLOW RATE AND QUALITY IS EXPLAINED AND ILLUSTRATED IN SOME CASES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

BLOWDOWN + *INSTRUMENT, FLOW + STEAM QUALITY + REACTOR, LWR + VOID FRACTION + *FLOW, TWO PHASE + *ANALYTICAL MODEL + ACCIDENT, LOSS OF COOLANT + FOREIGN EXCHANGE

134161
ROZENBERG J
EXAMINATION OF POSSIBLE RELATIONSHIPS BETWEEN WATER CHEMISTRY AND CLADDING BURST - IRENE CYCLE 05 (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
DRA/SAECMI/76/294/511/JR-FMC + FRRSR-97 +. 25 PPS, 8 TABS, 8 REFS, MAY 1976

DEPOSITS WERE FORMED ONLY ON THE PINS EXPOSED TO THE HIGHEST THERMAL FLUX. THE QUANTITY OF MATTER IN THESE DEPOSITS IS SMALL AND THEIR COMPOSITION IS VARIABLE. BASED CHIEFLY ON CARBON THEY ALSO CONTAIN ZIRCONIUM AND MOST ELEMENTS FOUND IN CORROSION PRODUCTS, TOGETHER WITH RELATIVELY HIGH CONCENTRATIONS OF CALCIUM AND ALUMINIUM, AND LASTLY WERE FOUND TO ADSORB LARGE QUANTITIES OF FISSION PRODUCTS. TO OBTAIN A BETTER ESTIMATE OF HOW THE QUALITY OF THE WATER AFFECTS THE MECHANISMS OF THIS KIND OF BURST A MORE FREQUENT EXAMINATION OF THE PINS SEEMS NECESSARY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CLADDING + FAILURE + COOLANT CHEMISTRY + CORROSION + FOREIGN EXCHANGE

2. GERMAN (FRG) LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE FEDERAL REPUBLIC OF GERMANY DURING THE FIRST HALF OF 1978 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

136289
CHRISTIAN H + ELFINGER FX
DETERMINATION OF WELD STRESSES WITH THE USE OF QUICKLY INSERTED RADIOGRAPHY - FINAL REPORT (IN GERMAN)
ALLIANZ-ZENTRUM FUR TECHNIK GMBH, F.R. GERMANY
RS 130 + GERRSR-190 +. 50 PPS, FIGS, 22 REFS, 1977

NO ENGLISH ABSTRACT PROVIDED AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*WELDS + STRESS + *STRESS ANALYSIS + WELDING + FOREIGN EXCHANGE

135308
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. FINAL RESEARCH REPORT D8 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF RS 50-32-D8 + GERRSR-153 +. APPROX. 110 PPS, FIGS, AUG. 1977

THE RESULTS OF EARLY TESTS IN PROJECT RS 50 INDICATE THAT THE COMPUTER CODES USED FOR THE DESIGN OF REACTOR CONTAINMENTS YIELD ESTIMATES WHICH ARE MUCH TOO HIGH. THIS FINAL REPORT INCORPORATES RESULTS OF SIMPLIFIED IDEALIZED EXPERIMENTS DESIGNED KTO PROVIDE DATA FOR THE DEVELOPMENT OF MORE ACCURATE COMPUTER CODES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EXPERIMENT + COMPUTER PROGRAM + CONTAINMENT DESIGN + REACTOR, LWR + *ACCIDENT, LOSS OF COOLANT + *CONTAINMENT R AND D + FOREIGN EXCHANGE

135307
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. QUICK LOOK REPORT EXPERIMENT D8 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-30-D8 + GERRSR-154 +. APPROX. 100 PPS, FIGS, JULY 1977

THE RESULTS OF EARLY TESTS IN PROJECT RS 50 INDICATE THAT THE COMPUTER CODES USED FOR THE DESIGN OF REACTOR CONTAINMENTS YIELD ESTIMATES WHICH ARE MUCH TOO HIGH. THIS REPORT IS A QUICK LOOK REPORT OF THE DATA GATHERED IN EXPERIMENT D8, A SIMPLIFIED IDEALIZED EXPERIMENT DESIGNED TO PROVIDE DATA FOR THE DEVELOPMENT OF MORE ACCURATE COMPUTER CODES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EXPERIMENT + COMPUTER PROGRAM + CONTAINMENT DESIGN + REACTOR, LWR + *ACCIDENT, LOSS OF COOLANT + *CONTAINMENT R AND D + FOREIGN EXCHANGE

135311
SKOUTAJAN R + BAUKAL W + KONIG R + WAGNER W
MEASUREMENT OF THE VISCOSITY OF SELECTED SUBSTANCES IN A CORE MELTDOWN (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-R-63.077-1 + GERRSR-170 +. 85 PPS, FIGS, REFS, NOV. 1977

THE OBJECTIVE OF PROJECT RS 214 IS TO CONDUCT THE PREPARATORY WORK NECESSARY FOR THE EXPERIMENTAL DETERMINATION OF THE VISCOSITY OF OXIDIC MELTS BY MEANS OF A HIGH-TEMPERATURE ROTARY VISCOMETER. THE INVESTIGATIONS WERE FOCUSED ON THE MIXTURES OF CORE MELT AND CONCRETE. FOR THOSE COMPONENTS OF THE VISCOMETER WHICH DURING A MEASUREMENT WILL COME INTO CONTACT WITH THE OXIDIC CORIUM-CONCRETE MIXTURE LED TO THE SELECTION OF TUNGSTEN. IN ADDITION, THE REPORT CONTAINS A DESCRIPTION OF PREPARATORY WORK NECESSARY FOR THE USE OF MEASURING SYSTEMS MADE OF TANTALUM, BORON NITRIDE, AND THORIUM OXIDE; ACCORDING TO SCHEDULE THIS WORK WAS POSTPONED IN FAVOR OF TUNGSTEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORE MELTDOWN + MEASUREMENT + CONCRETE + HIGH TEMPERATURE + TANTALUM + TUNGSTEN + FOREIGN EXCHANGE

132853
INVESTIGATION INTO THE PHENOMENA INVOLVED IN THE DEPRESSURIZATION OF WATER-COOLED REACTORS. EXPERIMENTS USING A STEEL VESSEL 11.2 M IN HEIGHT WITH INTERNALS. RESULTS OF THE FIRST PWR-EXPERIMENTS WITH INTERNALS (NOS. DWR1-DWR5) (IN GERMAN)
BATTELLE INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 0016B-10-1 + GERRSR0171 +. 133 PPS, 11 TABS, 52 FIGS, SEPT. 1977

THE REPORT DESCRIBES AND INTERPRETS THE EXPERIMENTAL DATA OBTAINED IN A FIRST SERIES OF BLOWDOWN EXPERIMENTS UNDER PRESSURIZED WATER REACTOR CONDITIONS USING A MODEL REACTOR PRESSURE VESSEL EQUIPPED WITH REPRESENTATIVE INTERNALS. THE SPECIFIC OBJECTIVE OF THESE EXPERIMENTS WAS TO INVESTIGATE THE DISCHARGE PROCESS IN THE RUPTURED NOZZLE AND THE LOADS ON AND STRAINS IN THE INTERNALS. ON THE BASIS OF THE DATA, MEASURED WITH ABOUT 180 MEASURING SYSTEMS INSTALLED WITHIN

132853 *CONTINUED*

AND ON THE PRESSURE VESSEL. THE THERMOHYDRAULIC PHENOMENA OCCURRING IN THE RUPTURED NOZZLE AND INSIDE THE VESSEL AND THE STRUCTURAL BEHAVIOR OF THE INTERNALS WERE STUDIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*BLOWDOWN + *ACCIDENT. LOSS OF COOLANT + NOZZLE + REACTOR. PWR + EXPERIMENT + TEST. PRESSURE VESSEL + FOREIGN EXCHANGE

137468
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS QUICK LOOK REPORT EXPERIMENT D9 (IN GERMAN)
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
BF-RS 50-30-D9 + GERRSR-174 +. APPROX. 150 PPS. TABS. FIGS. NOV. 1977

QUICK LOOK REPORT OF EXPERIMENT DESIGNED TO PROVIDE EXPERIMENTAL DATA FOR THE IMPROVEMENT OF EXISTING COMPUTER CODES. IMPORTANT RESULTS OF EXPERIMENT D9 ARE: CHANNEL FLOW IS HIGHER THAN ORIFICE FLOW OF EQUAL DIAMETER; USING FLOW DATA OF FROSSEL IN THE CALCULATIONS GOOD AGREEMENT WITH EXPERIMENTAL DATA RESULTS. QUASI-STATIONARY AND NON-STATIONARY CALCULATIONS OF CHANNEL FLOW SHOWED LITTLE DIFFERENCES IN THE RESULTS. IN MARKED DIFFERENCE TO THE RESULTS OF THE CALCULATIONS AND TO PREVIOUS EXPERIMENTS. THERE EXISTED ONLY A SMALL DIFFERENCE BETWEEN THE PRESSURES IN THE COMPARTMENTS R8. R7. R9 FOLLOWING THE BREAK COMPARTMENT; MODELING THESE THREE COMPARTMENTS AS ONE NODAL ZONE RESULTED IN AN IMPROVED AGREEMENT BETWEEN CALCULATED AND EXPERIMENTAL DATA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*ACCIDENT. LOSS OF COOLANT + COMPUTER PROGRAM + HEAT TRANSFER EXPERIMENT + *FLOW THEORY AND EXPERIMENTS + FLOW. TWO PHASE + COMPARTMENT + *CONTAINMENT ANALYSIS + REACTOR. LWR + *BLOWDOWN + COMPARISON. THEORY AND EXPERIENCE + FOREIGN EXCHANGE

135417
LOCA EXPERIMENTS WITH A PWR MULTI-COMPARTMENT MODEL CONTAINMENT
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
BF-RS 50-62-5 + GERRSR-175 +. 23 PPS. 12 FIGS. 5 REFS. OCT. 1977

THE LOADS ACTING ON A CONTAINMENT DURING A LOCA (JET FORCES. DIFFERENTIAL PRESSURES. MAXIMUM PRESSURE) WERE EXPERIMENTALLY INVESTIGATED IN A MODEL CONTAINMENT OF 12 M IN DIAMETER. TO VERIFY AND TO IMPROVE THE CALCULATIONAL METHODS THE EXPERIMENTAL RESULTS ARE COMPARED WITH MODEL CALCULATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

CONTAINMENT + TESTING + ANALYTICAL MODEL + COMPARISON + FOREIGN EXCHANGE + ACCIDENT. LOSS OF COOLANT

137477
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. SUPPLEMENTAL EXPERIMENTAL DOCUMENTATION D3 (IN GERMAN)
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
BF-RS 50-32-D3 + GERRSR-180 +. APPROX. 150 PPS. TABS. FIGS. OCT. 1977

SUPPLEMENTAL REPORT OF EXPERIMENT DESIGNED TO PROVIDE EXPERIMENTAL DATA FOR THE IMPROVEMENT OF EXISTING COMPUTER CODES. THIS REPORT INCLUDES A COMPARISON OF TEST D3 RESULTS WITH THE EXPERIMENTS IN THE D SERIES. IN ADDITION. PRE- AND POST-EXPERIMENT CALCULATIONS WERE PERFORMED AND COMPARED FOR TEST D3.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*ACCIDENT. LOSS OF COOLANT + COMPUTER PROGRAM + HEAT TRANSFER EXPERIMENT + *FLOW THEORY AND EXPERIMENTS + FLOW. TWO PHASE + COMPARTMENT + *CONTAINMENT ANALYSIS + REACTOR. LWR + *BLOWDOWN + COMPARISON. THEORY AND EXPERIENCE + FOREIGN EXCHANGE

136290
JAX P
PRELIMINARY EXPERIMENT ON INSPECTION FOR LEAKAGE (CRACKS) WITH THE USE OF ACOUSTICS EMISSION ANALYSIS - FINAL REPORT (IN GERMAN)
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
BF-R-62.944-1 + GERRSR-191 +. 100 PPS. 9 TABS. 33 FIGS. 10 REFS. MARCH 1977

NO ENGLISH ABSTRACT PROVIDED AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

GERMANY + EXPERIMENT + *LEAK DETECTION + *ACOUSTICS + SURVEILLANCE PROGRAM + FOREIGN EXCHANGE

132212
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS COMPLETE DOCUMENTATION EXPERIMENT D9 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-32-D9 + GERRSR-207 +. APPROX. 130 PPS. TABS. FIGS. NOV. 1977

COMPLETE REPORT OF EXPERIMENT DESIGNED TO PROVIDE EXPERIMENTAL DATA FOR THE IMPROVEMENT OF EXISTING COMPUTER CODES. IMPORTANT RESULTS OF EXPERIMENT D9 ARE: CHANNEL FLOW IS HIGHER THAN ORIFICE FLOW OF EQUAL DIAMETER; USING FLOW DATA OF FROSSEL IN THE CALCULATIONS GOOD AGREEMENT WITH EXPERIMENTAL DATA RESULTS. QUASI-STATIONARY AND NON-STATIONARY CALCULATIONS OF CHANNEL FLOW SHOWED LITTLE DIFFERENCES IN THE RESULTS. IN MARKED DIFFERENCE TO THE RESULTS OF THE CALCULATIONS AND TO PREVIOUS EXPERIMENTS, THERE EXISTED ONLY A SMALL DIFFERENCE BETWEEN THE PRESSURES IN THE COMPARTMENTS R6, R7, R9 FOLLOWING THE BREAK COMPARTMENT; MODELING THESE THREE COMPARTMENTS AS ONE NODAL ZONE RESULTED IN AN IMPROVED AGREEMENT BETWEEN CALCULATED AND EXPERIMENTAL DATA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ACCIDENT, LOSS OF COOLANT + COMPUTER PROGRAM + HEAT TRANSFER EXPERIMENT + *FLOW THEORY AND EXPERIMENTS + FLOW, TWO PHASE + COMPARTMENT + *CONTAINMENT ANALYSIS + REACTOR, LWR + *BLOWDOWN + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE

132213
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS QUICK LOOK REPORT EXPERIMENT D11 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-30-D11 + GERRSR-208 +. APPROX. 150 PPS. TABS. FIGS. JAN. 1978

QUICK LOOK REPORT OF EXPERIMENT DESIGNED TO PROVIDE EXPERIMENTAL DATA FOR THE IMPROVEMENT OF EXISTING COMPUTER CODES. IMPORTANT RESULTS OF EXPERIMENT D11 ARE: COMPARTMENT REGIONS, WHICH ARE NOT ON THE DIRECT FLOW-PATH OF THE BREAK AND FOLLOWING COMPARTMENTS ("DEAD END"), ARE PURGED SLOWLY AND NOT COMPLETELY. THUS STRONG INHOMOGENITIES IN THE COMPOSITION OF THE VAPOR/AIR MIXTURE HAVE DEVELOPED. IN THE BREAK ROOM AND FOLLOWING COMPARTMENT WITH TRANSVERSAL FLOW, MARKED PRESSURE WAVES OCCUR WITH A WAVE LENGTH CORRESPONDING TO TWICE THE LENGTH OF THE COMPARTMENT. THEY MAKE AN IMPORTANT CONTRIBUTION TO THE MEASURED MAXIMUM PRESSURE DIFFERENCE. GOOD AGREEMENT BETWEEN MEASURED AND CALCULATED (ZOCO 6 CALCULATION) PRESSURE DATA IN THE SHORT TERM BEHAVIOR HAS BEEN ACHIEVED USING DIFFERENT ASSUMPTIONS FOR THE HEAT TRANSFER COEFFICIENT ALPHA(SUB W) AND FLOW COEFFICIENT ALPHA(SUB D).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ACCIDENT, LOSS OF COOLANT + COMPUTER PROGRAM + HEAT TRANSFER EXPERIMENT + FLOW, TWO PHASE + COMPARTMENT + *CONTAINMENT ANALYSIS + REACTOR, LWR + *BLOWDOWN + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE + *FLOW THEORY AND EXPERIMENTS

132788
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. COMPLETE DOCUMENTATION EXPERIMENT D11 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-32-D11 + GERRSR-209 +. APPROX. 125 PPS. TABS. FIGS. FEB. 1978

COMPLETE REPORT OF EXPERIMENT DESIGNED TO PROVIDE EXPERIMENTAL DATA FOR THE IMPROVEMENT OF EXISTING COMPUTER CODES. IMPORTANT RESULTS OF EXPERIMENT D11 ARE: COMPARTMENT REGIONS, WHICH ARE NOT ON THE DIRECT FLOW-PATH OF THE BREAK AND FOLLOWING COMPARTMENTS ("DEAD END"), ARE PURGED SLOWLY AND NOT COMPLETELY. THUS STRONG INHOMOGENITIES IN THE COMPOSITION OF THE VAPOR/AIR MIXTURE HAVE DEVELOPED. IN THE BREAK ROOM AND FOLLOWING COMPARTMENT WITH TRANSVERSAL FLOW, MARKED PRESSURE WAVES OCCUR WITH A WAVE LENGTH CORRESPONDING TO TWICE THE LENGTH OF THE COMPARTMENT. THEY MAKE AN IMPORTANT CONTRIBUTION TO THE MEASURED MAXIMUM PRESSURE DIFFERENCE. GOOD AGREEMENT BETWEEN MEASURED AND CALCULATED (ZOCO 6 CALCULATION) PRESSURE DATA IN THE SHORT TERM BEHAVIOR HAS BEEN ACHIEVED USING DIFFERENT ASSUMPTIONS FOR THE HEAT TRANSFER COEFFICIENT ALPHA(SUB W) AND FLOW COEFFICIENT ALPHA(SUB D).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF COOLANT + COMPUTER PROGRAM + HEAT TRANSFER EXPERIMENT + FLOW THEORY AND EXPERIMENTS + FLOW, TWO PHASE + COMPARTMENT + *CONTAINMENT ANALYSIS + REACTOR, LWR + BLOWDOWN + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE

136179
SAHM A
PROPAGATION OF SIMULTANEOUS ACOUSTIC EMISSION - IMPULSES IN THICK-WALLED STRUCTURAL MEMBERS
BATTELLE-INSTITUTE E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-R-62.945-1 + GERRSR-211 +. 101 PPS, 4 TABS, 41 FIGS, 42 REFS, DEC. 1977

INCLUDED AMONG TOPICS DISCUSSED IS MEASUREMENT OF ACOUSTIC EMISSIONS BY FRACTURE MECHANICS PROBES-FATIGUE FRACTURE.

136179 *CONTINUED*
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *ACOUSTICS + *STRUCTURAL ANALYSIS, DYNAMIC + MEASUREMENT + FAILURE, FATIGUE + FATIGUE + CRACK + FOREIGN EXCHANGE

137455
BOHM B
PRESSURE AND TEMPERATURE HISTORIES IN BLOWDOWN EXPERIMENTS IN A MODEL CONTAINMENT (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-62-2 + GERRSR-215 +. 27 PPS. 1 TAB. 12 FIGS. OCT. 1975

PRESSURE AND TEMPERATURE HISTORIES OCCURING UNDER PWR CONDITIONS DURING THE SIMULATION OF A ONE-END RUPTURE OF THE MAIN COOLANT PIPE IN A MODEL CONTAINMENT ARE SHOWN. THE EXPERIMENTAL FACILITY, WHICH IS QUARTER-SCALE MODEL OF A CONTAINMENT OF A 1200 MWE PWR NUCLEAR POWER PLANT, IS BRIEFLY DESCRIBED. THE PRESENT REPORT COVERS THE PRESSURE AND TEMPERATURE HISTORIES DETERMINED WITHIN THE MODEL CONTAINMENT DURING THE PHASES OF PRESSURE RISE AND PRESSURE DECREASE AND A DISCUSSION OF RELEVANT PHENOMENA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + *BLOWDOWN + *EXPERIMENT + *CONTAINMENT + *MODEL + FOREIGN EXCHANGE

136218
COMPLETION OF R & D WORK ON DEPRESSURIZATION DURING A BLOWDOWN IN A WATER-COOLED REACTOR - THE EXPERIMENTS USED A STEEL VESSEL 11.2 METERS IN HEIGHT WITH INTERNALS (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS00168-32-5 + GERRSR-219 +. APPROX. 75 PPS. FIGS. AUG. 1975

NO FURTHER TRANSLATION WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + R AND D PROGRAM + *BLOWDOWN + *PRESSURE TRANSIENT + ACCIDENT, LOSS OF COOLANT + REACTOR, LWR + FOREIGN EXCHANGE

136194
HEIDE W
TRANSMISSION PROPERTIES OF ACOUSTIC WAVES, RESEARCH PROJECT RS 31/3 RESEARCH STUDIES ON THE CONTINUOUS INSPECTION FOR FLAWS IN PRESSURE VESSELS, PART IV (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
RS 31/3 + GERRSR-220 +. 36 PPS. 13 FIGS. 10 REFS. SEPT. 1974

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

R AND D PROGRAM + ACOUSTICS + *SURVEILLANCE PROGRAM + *PRESSURE VESSELS + FLAW + FOREIGN EXCHANGE + GERMANY

136199
VON KLOT R + SAHM A
ENHANCEMENT OF THE DETECTION PROBABILITY OF FLAWS THROUGH IMPROVEMENT OF THE SIGNAL-TO-NOISE RATIO USING ULTRASONIC TESTING (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-R-62.948-1 + GERRSR-221 +. 37 PPS. 1 TAB. 18 FIGS. AUG. 1976

NO ENGLISH ABSTRACT WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *ULTRASONICS + TESTING + *FLAW + INSTRUMENT, SURVEILLANCE + NOISE ANALYSIS + SENSITIVITY ANALYSIS + FOREIGN EXCHANGE

138036
HAMPEL G + SOFFKE W + SCHULTZ H
IMPROVEMENT OF A DRAG BODY FOR THE MASS FLOW MEASUREMENT IN BLOWDOWN EXPERIMENTS, PART 1 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-R-62.519-1 + RS-147 + GERRSR-231 +. 110 PPS. FIGS. REFS. APRIL 1978

THE PRESENT PART I OF THE REPORT ON PROJECT RS 147 COVERS THE DEVELOPMENT OF FOUR PROTOTYPE DRAG BODIES WITH DIFFERENT MEASURING RANGES WHICH ARE TO BE USED FOR MASS FLOW MEASUREMENTS ON TRANSIENT TWO-PHASE FLOWS WITHIN THE EXPERIMENTAL BLOWDOWN PROGRAM RS 109 (CONDUCTED AT EURATOM/ISPRA). THE REPORT DEALS WITH THE DESIGN CALCULATIONS AND SPECIFICALLY WITH THE THEORETICAL AND EXPERIMENTAL INVESTIGATIONS INTO THE DYNAMIC BEHAVIOR OF DRAG BODIES AIMED AT

138036 *CONTINUED*

OPTIMIZING THE DESIGN PARAMETERS. IN ADDITION, THE DESIGN OF THE DRAG BODIES, THE TRANSDUCER CIRCUITRY AND THE FUNCTIONAL TESTS CARRIED OUT WITH THE PROTOTYPES ARE DESCRIBED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*FLOW, TWO PHASE + BLOWDOWN + ACCIDENT, LOSS OF COOLANT + MEASUREMENT + FLOW THEORY AND EXPERIMENTS + *SENSORS, FLOW + COMPUTER PROGRAM + ANALYTICAL MODEL + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE

132789

INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS QUICK LOOK REPORT EXPERIMENT D10 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-30-D10 + GERRSR-232 +. APPROX. 125 PPS, TABS, FIGS, APRIL 1976

QUICK LOOK REPORT OF EXPERIMENT DESIGNED TO PROVIDE EXPERIMENTAL DATA FOR THE IMPROVEMENT OF EXISTING COMPUTER CODES. MAIN RESULTS OF THE EXPERIMENT D10 ARE: THE LARGER OVERFLOW CROSS SECTIONS OF THE COMPARTMENT R8 (IN COMPARISON TO PREVIOUS EXPERIMENTS) EFFECTED LOW DIFFERENTIAL PRESSURE PEAKS (LESS THAN 0.1 BAR) AND A QUICK PRESSURE EQUALIZATION BETWEEN FOLLOWING COMPARTMENTS TO A RELATIVELY LOW LEVEL PRIMARILY DETERMINED BY THE EXTENDED DOME R9. THE PRESSURE DIFFERENCE BETWEEN BREAK COMPARTMENT AND FIRST FOLLOWING COMPARTMENT R8 IS RELATIVELY HIGH, IN COMPARISON TO PREVIOUS CORRESPONDING EXPERIMENTS, DUE TO THE LOW PRESSURE LEVEL IN THE COMPARTMENT R8.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ACCIDENT, LOSS OF COOLANT + COMPUTER PROGRAM + HEAT TRANSFER EXPERIMENT + *FLOW THEORY AND EXPERIMENTS + FLOW, TWO PHASE + COMPARTMENT + *CONTAINMENT ANALYSIS + REACTOR, LWR + *BLOWDOWN + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE

137458

RELIABILITY ANALYSIS OF THE BWR-6 EMERGENCY COOLING SYSTEM (IN GERMAN)
B.U.C.K., F.R. GERMANY
RS-134 + GERRSR-214 +. 84 PPS, 17 TABS, 14 FIGS, AUG. 25, 1975

THE RELIABILITY PARAMETERS OF THE EMERGENCY CORE COOLING SYSTEM OF THE BWR/6 ARE DETERMINED FOLLOWING A LARGE LOSS-OF-COOLANT ACCIDENT. THE INVESTIGATED TIME PERIOD IS SUBDIVIDED INTO TWO DRIFT PHASES: PHASE 1: THE STANDBY PERIOD AND THE FLOODING OF THE CORE - PHASE 2: THE PERIOD OF THE LONG-TIME HEAT-REMOVAL. THE CALCULATIONS ARE PERFORMED WITH THE MULTI-PHASE SIMULATION PROGRAM SYG/1. SINCE CALCULATIONS ARE CARRIED OUT IN ONE RUN FOR THE WHOLE SYSTEM, IT IS UNNECESSARY TO DISTINGUISH BETWEEN FAULT TREES AND EVENT TREES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF COOLANT + *EMERGENCY COOLING SYSTEM + REACTOR, BWR + ANALYTICAL TECHNIQUE + COMPUTER PROGRAM + *SIMULATION + FAULT TREE ANALYSIS + *RELIABILITY ANALYSIS + FOREIGN EXCHANGE

136203

DROGE KH
ELECTRICAL ANTHROPOMORPHOUS TELEMANIPULATING FOR NUCLEAR PLANTS (IN GERMAN)
ELEKTRONIK FUR RAUMFAHRT UND ATOMTECHNIK, F.R. GERMANY
BMFT-FB RS 21/4 + GERRSR-216 +. 47 PPS, FIGS, 43 REFS, JUNE 1974

DESCRIBES A MASTER-SLAVE MANIPULATOR SYSTEM WITH ANTHROPOMORPHOUS CINEMATIC. THE SYSTEM IS REMOTELY CONTROLLED, CORRESPONDING TO THE NUCLEAR APPLICATIONS. SUBSYSTEMS - LIKE STEREO TELEVISION SYSTEM, FORCE-FEEDBACK, NOISE-TRANSMISSION - DELIVER INFORMATION TO THE OPERATOR ABOUT THE MACHINE AND ITS SURROUNDING. THE LOCOMATION SYSTEM IS A THREE WHEELED VEHICLE. TRANSPORTATION IS DONE BY A SMALL ESPECIALLY EQUIPPED VAN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*REMOTE MANIPULATING AND VIEWING + TELEVISION + FOREIGN EXCHANGE

132555

LANGHEIM H
RESEARCH ON THE STOPPING POWER OF MATERIALS AND STRUCTURES IN REACTOR BUILDINGS AGAINST IMPINGING FRAGMENTS AND PROJECTILES AS A FUNCTION OF THEIR MASS AND VELOCITY (IN GERMAN)
ERNST-MACH-INSTITUT, GERMANY
RS 102-07-5 + EMI-E 14/77 + GERRSR-172 +. 63 PPS, 8 TABS, 40 FIGS, 5 REFS, JUNE 1977

THE INFLUENCE OF REINFORCEMENTS ON THE LOCAL RESISTANCE OF CONCRETE SLABS WAS INVESTIGATED. CYLINDRICAL SOLID-STEEL PROJECTILES WERE FIRED INTO SQUARE CONCRETE SLABS HAVING 4 DIFFERENT REINFORCEMENTS AT VELOCITIES FROM 150 TO 430 M/S. THE PICTURES OF THE DAMAGE ON THE REAR SIDE OF THE SLABS UNDER ULTIMATE LOAD SHOW THE INFLUENCE OF THE REINFORCEMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

132555 *CONTINUED*
CONCRETE + CONTAINMENT INTEGRITY + DAMAGE + *MISSILE GENERATION AND PROTECTION + GERMANY + FOREIGN EXCHANGE

137744
LANGHEIM H
BEHAVIOR OF REACTOR SPECIFIC MATERIALS AND COMPONENT PARTS AT THE IMPACT OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS AND VELOCITY (IN GERMAN)
ERNST-MACH-INSTITUT. F.R. GERMANY
EMI-E 18/77 + RS-102-07-6 + GERRSR-235 +. 80 PPS. FIGS. AUG. 1977

OBJECT OF TESTS WAS THE RESISTANCE BEHAVIOR OF STEEL AND REINFORCED CONCRETE SLABS DURING SHORT-TIME BALLISTIC LOADS. PENETRATION CURVES WERE PLOTTED. TO DETERMINE PENETRATION DEPTHS FOR SOLID STEEL CYLINDERS WITH L/D = 1 AND L/D = 2 AND IMPACT VELOCITIES FROM 150 TO 1750 M/S IN CUBICALLY REINFORCED CONCRETE TARGETS. IN THE TEST SERIES "STEEL TARGETS" LIMIT SLAB THICKNESSES OF REACTOR STEELS 22 NI MO CR 37 WERE DETERMINED UNDER VARIOUS LOADS. THE EXPERIMENTAL PENETRATION DEPTHS FOR REINFORCED CONCRETE AND THE LIMIT SLAB THICKNESSES OF STEEL WERE COMPARED WITH THE CALCULATED VALUES USING EMPIRICAL FORMULAE. THERE DOES NOT EXIST A FORMULA PERMITTING SATISFACTORY THEORETICAL PREDICTIONS OF THE DAMAGES UNDER SHORT TIME DYNAMIC LOADS OVER A WIDE PROJECTILE AND TARGET RANGE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

GERMANY + MISSILE GENERATION AND PROTECTION + IMPACT SHOCK + STEEL + CONCRETE. REINFORCED + DAMAGE + FOREIGN EXCHANGE

137598
LOFFLER E
LITERATURE SEARCH II. ON THE RESEARCH PROJECT "DEFORMATION OF THE REACTOR BUILDING BY A PRESSURE WAVE
ERNST-INSTITUT. F.R. GERMANY
EMI-E 23/75 + RS-102-09-3 + GERRSR-236 +. 15 PPS. 11 TABS. 3 FIGS. 9 REFS. DEC. 1975

NO ENGLISH ABSTRACT WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

BIBLIOGRAPHY + *REACTOR + *BUILDING + *DEFORMATION + SHOCK WAVE + PRESSURE. EXTERNAL + CONTAINMENT INTEGRITY + FOREIGN EXCHANGE

137613
OSTERHOLT K
BEHAVIOR OF PRIMARY PUMP IN ONE AND TWO PHASE FLOW - FEASIBILITY AND EXTENT OF THE COMPLETION OF THE RESEARCH PROGRAM (IN GERMAN)
GESELLSCHAFT FUR KERNENERGIEVERWERTUNG. F.R. GERMANY
GKSS 77/I/19 + GERRSR-233 +. 115 PPS. FIGS. 1977

DESCRIBES THE RESEARCH FACILITY USED FOR STUDYING AND DETERMINING THE STATIONARY BEHAVIOR OF A REACTOR PRIMARY PUMP IN ONE AND TWO PHASE FLOW CONDITIONS. THE FEASIBILITY AND EXTENT OF COMPLETION OF THE PROGRAM ARE DISCUSSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

GERMANY + *PUMPS + MAIN COOLING SYSTEM + *HYDRAULIC ANALYSIS + FLOW. TWO PHASE + R AND D PROGRAM + FOREIGN EXCHANGE

137614
BIEMANN P + BOTTCHER H + DOOSE R
BEHAVIOR OF REACTOR PRIMARY PUMP IN ONE- AND TWO-PHASE FLOW (IN GERMAN)
GESELLSCHAFT FUR KERNENERGIEVERWERTUNG. F.R. GERMANY
GKSS 77/I/22 + GERRSR-234 +. APPROX. 85 PPS. FIGS. 1977

PRESENTS THE DESIGN FOR THE RESEARCH FACILITY TO BE USED FOR THE DETERMINATION OF THE STATIONARY BEHAVIOR OF A REACTOR PUMP IN ONE- AND TWO-PHASE FLOW.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*DESIGN CRITERIA + *REACTOR TEST FACILITY + *PUMPS + MAIN COOLING SYSTEM + HYDRAULIC ANALYSIS + FLOW. TWO PHASE + FOREIGN EXCHANGE

132552
BACHNER D + MULLER W
RADIATION EXPOSURE AND RADIATION RISK THROUGH INHALATION OR INGESTION OF ACTINIDES (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH. F.R. GERMANY
GRS-1 + GERRSR-155 +. 124 PPS. TABS. FIGS. 72 REFS. AUG. 1977

132552 *CONTINUED*

ON THE BASIS OF THE LUNG MODEL FOLLOWING THE ICRP RECOMMENDATIONS, DOSE COMMITMENTS AND DOSE FACTORS HAVE BEEN CALCULATED FOR THE LUNG, LYMPHATIC TISSUE, BONE AND LIVER DUE TO SINGLE UPTAKE OF RADIOACTIVE NUCLIDES BY INHALATION OR INGESTION. THESE CALCULATIONS HAVE BEEN PERFORMED FOR THE MOST IMPORTANT ISOTOPES OF THORIUM, URANIUM, NEPTUNIUM, PLUTONIUM, AMERICIUM, CURIUM AND FOR MIXTURES OF PLUTONIUM. ANALOGIES BETWEEN DOSE AND DOSE RATE DUE TO SINGLE AND CONTINUOUS INTAKE OF RADIOACTIVE NUCLIDES ARE DISCUSSED. FINALLY ESTIMATIONS OF THE COLLECTIVE AND INDIVIDUAL RISK ARE MADE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*DOSE CALCULATION, INTERNAL + ANALYTICAL MODEL + POPULATION EXPOSURE + GERMANY + LUNG + FOREIGN EXCHANGE

133734
KUNTZE WM
ANALYSIS OF THE DYNAMIC BEHAVIOR OF NUCLEAR PLANTS IN CASE OF EXTERNAL INFLUENCES CALCULATION OF THE EARTHQUAKE LOADING OF THE REACTOR BUILDING SWR-72 BY THE FINITE-ELEMENT PROGRAM SYSTEM ASKA. DOCUMENTATION (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-2 + GERRSR-156 +. 98 PPS, 39 FIGS, 11 REFS, NOV, 1977

FROM AVAILABLE RECORDINGS, DISPLACEMENT TIME HISTORIES FOR THREE ORTHOGONAL DIRECTIONS (TWO IN HORIZONTAL DIRECTION AND ONE IN VERTICAL DIRECTION) OF THE SAN FERNANDO EARTHQUAKE WERE SELECTED. THE MODEL OF THE REACTOR BUILDING IS EXCITED AT ITS FOOTING; THE SOIL IS NOT TAKEN INTO ACCOUNT. THE REACTOR BUILDING OF SWR 72 IS MODELLED WITH FINITE ELEMENTS. PROVIDED THE VERY HEAVY AND THICK FOOTING REACTS AS A RIGID BODY, THE MODEL OF THE WHOE REACTOR BUILDING CAN BE DECOMPOSED IN FOUR DYNAMICALLY INDEPENDENT SUBMODELS, THEREBY DISREGARDING THE FOOTING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EARTHQUAKE + STRUCTURAL ANALYSIS, DYNAMIC + POWER PLANT, NUCLEAR + GERMANY + MATHEMATICAL TREATMENT + FOREIGN EXCHANGE

132654
PROGRESS REPORT. LIST OF REACTOR SAFETY RESEARCH REPORTS FROM BMFT AND USNRC DURING THE PERIOD FROM APRIL 1 TO JUNE 30, 1977
GESELLSCHAFT FUR REAKTORSICHERHEIT MBH, F.R. GERMANY
GRS-F-43 + GERRSR-157 +. 40 PPS, AUG, 1977

THIS LIST REVIEWS REPORTS CONCERNING SPECIAL PROBLEMS IN THE FIELD OF REACTOR SAFETY RESEARCH WHICH HAVE BEEN COLLECTED IN THE GESELLSCHAFT FUR REAKTORSICHERHEIT. THE LIST IS IN THE FOLLOWING ORDER: COUNTRY OF ORIGIN, PROBLEM AREA CONCERNED, ACCORDING TO THE REACTOR SFAETY RESEARCH PROGRAM OF BMFT, REPORTING ORGANIZATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

BIBLIOGRAPHY + R AND D PROGRAM + SAFETY PROGRAM + GERMANY + ENGINEERED SAFETY FEATURE + FOREIGN EXCHANGE

132846
RINGER FJ
DETERMINATION OF THE FLOW FORCES ON THE LOBI CIRCUIT. INTERPRETATION OF THE CALCULATIONS (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-22 + GERRSR-159 +. 28 PPS, 4 FIGS, 3 REFS, JULY 1977

AS PART OF THE REACTOR SAFETY EXPERIMENT RS 109, THE STRUCTURE DYNAMIC BEHAVIOR OF THE LOBI (LOOP-BLOWDOWN INVESTIGATION) PILOT PLANT WAS INVESTIGATED AT THE EURATOM ISPRA LABORATORY. THE WORK INCLUDED DETERMINATION OF THE FLOW FORCES TO WHICH THE STRUCTURE IS EXPOSED DURING A BLOWDOWN. THE STRUCTURE WAS EXPOSED TO A STATIC INTERNAL PRESSURE, ADDITIONAL FORCES WERE CREATED BY FRICTION AND BY THE DEVIATIONS AND TURNS OF THE CURRENT. THE CALCULATION REVEALED THAT THE FLOW FORCE REACHED A MAXIMUM DURING THE FIRST PHASE OF THE BLOWDOWN. DURING THE COURSE OF THE BLOWDOWN THE FLOW FORCE EXHIBITED MARKED OSCILLATIONS WHICH MIGHT BE RELATED WITH THE PROPAGATION OF PRESSURE WAVES. RESULTS OF THE CALCULATIONS ARE SHOWN GRAPHICALLY IN THE APPENDIX; THEY REPRESENT THE BEHAVIOR DURING THE FIRST 0.125 SECOND OF THE BLOWDOWN. THE RESULTS ARE ALSO RECORDED FOR THE SAME TIME PERIOD ON A MAGNETIC TAPE AT A RATE OF ABOUT 3 X 10(EXP -4) SEC.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*BLOWDOWN + *STRUCTURAL ANALYSIS, DYNAMIC + *PRESSURE PULSE + COMPUTER PROGRAM + FOREIGN EXCHANGE

135038
RINGER FJ
DETERMINATION OF THE FLOW FORCES ON THE LOBI CIRCUIT. INTERPRETATION OF THE CALCULATIONS - APPENDIX (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-22 + GERRSR-160 +. APPROX. 175 PPS, FIGS, JULY 1977

136038 *CONTINUED*

AS PART OF THE REACTOR SAFETY EXPERIMENT RS 109, THE STRUCTURE DYNAMIC BEHAVIOR OF THE LOBI (LOOP-BLOWDOWN INVESTIGATION) PILOT PLANT WAS INVESTIGATED AT THE EURATOM ISPRA LABORATORY. THE WORK INCLUDED DETERMINATION OF THE FLOW FORCES TO WHICH THE STRUCTURE IS EXPOSED DURING A BLOWDOWN. THE STRUCTURE WAS EXPOSED TO A STATIC INTERNAL PRESSURE. ADDITIONAL FORCES WERE CREATED BY FRICTION AND BY THE DEVIATIONS AND TURNS OF THE CURRENT. THE CALCULATION REVEALED THAT THE FLOW FORCE REACHED A MAXIMUM DURING THE FIRST PHASE OF THE BLOWDOWN. DURING THE COURSE OF THE BLOWDOWN THE FLOW FORCE EXHIBITED MARKED OSCILLATIONS WHICH MIGHT BE RELATED WITH THE PROPAGATION OF PRESSURE WAVES. RESULTS OF THE CALCULATIONS ARE SHOWN GRAPHICALLY IN THE APPENDIX.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*REACTOR TEST FACILITY + *STRUCTURAL ANALYSIS, DYNAMIC + *THERMAL HYDRAULIC ANALYSIS + BLOWDOWN + PRESSURE, INTERNAL + FLOW + SHOCK WAVE + MATHEMATICAL TREATMENT + FOREIGN EXCHANGE

135412
ULLRICH R + NOLTE W
PRETEST PREDICTIONS OF THE NONNUCLEAR LOFT EXPERIMENT L1-4 (U.S. - STANDARD PROBLEM NO. 7) (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-63 + GERRSR-176 +. 209 PPS, TABS, FIGS, NOV. 1977

CONTAINS THE RESULTS OF PRETEST PREDICTIONS DONE BY BLOWDOWN CODE RELAP 4/GRS. RESULTS OF LOFT EXPERIMENT L1-4 PRETEST PREDICTIONS SUMMARIZED AS FOLLOWS: 1. ANALYSIS WAS SUBDIVIDED INTO A 51 VOLUME BLOWDOWN CALCULATION AND 2 VOLUME REFILL/REFLOOD CALCULATION, BECAUSE OF WATER PACKING PROBLEMS AFTER ECC INJECTION. 2. CALCULATIONS AGREE WELL WITH FIRST EXPERIMENTAL DATA. ACCURACY OF CALCULATIONS IMPROVED OVER L1-2 ANALYSIS. 3. PROGRAM HAS BEEN UPDATED TO AVOID WATER PACKING PROBLEMS. NEW VERSION SHOWS IMPROVEMENT OF RESULTS FOR SINGLE TEST EXAMPLE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ANALYTICAL MODEL + *COMPUTER PROGRAM + REACTOR, PWR + *ACCIDENT, LOSS OF COOLANT + *BLOWDOWN + CORE REFLOODING + FOREIGN EXCHANGE

135875
REPORT ON THE RESEARCH PROGRAM OF THE BMFT IN THE AREA OF REACTOR SAFETY, REPORTING PERIOD APRIL 1-JUNE 30, 1977 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-F-45 + GERRSR-183 +. 491 PPS, NOV, 1977

INVESTIGATIONS ON THE SAFETY OF LIGHT WATER REACTORS AS WELL AS ON THE SAFETY OF ADVANCED REACTORS BEING PERFORMED IN THE FRAMEWORK OF THE RESEARCH PROGRAM REACTOR SAFETY ARE SPONSORED BY THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY. OBJECTIVES OF THIS PROGRAM ARE TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*REACTOR, LWR + REACTOR, PWR + REACTOR, BWR + *SAFETY PROGRAM + SAFETY ANALYSIS + *R AND D PROGRAM + SAFETY MARGIN + TECHNOLOGY + GERMANY + FOREIGN EXCHANGE

136284
APEL G
COMPILATION OF ABNORMAL EVENTS WHICH OCCURRED WORLDWIDE IN POWER, RESEARCH, AND DEVELOPMENT REACTORS IN 1969 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-7 + GERRSR-184 +. 139 PPS, TABS, FIGS, DEC, 1977

THESE COMPILATIONS ARE BASED ON SCIENTIFIC AND TECHNICAL PUBLICATIONS, LIKE PERIODICALS, REPORTS AND NEW SERVICES. THE INDIVIDUAL EVENTS WERE GROUPED ACCORDING TO THE INSTALLATION AFFECTED, I.E., WHETHER POWER REACTORS, EXPERIMENTAL REACTORS, OR RESEARCH REACTORS. HOWEVER, ONLY IN THE UNITED STATES WAS THE REPORTING SYSTEM COMPLETE; IN ALL OTHER COUNTRIES, APART FROM THE REPORTING OF SERIOUS EVENTS, IT WAS MORE OR LESS LEFT TO THE DISCRETION OF THE UTILITY TO DETERMINE THE METHOD AND THE EXTENT OF THE REPORTING OF ABNORMAL EVENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

POWER PLANT, NUCLEAR + *INCIDENT COMPILATION + OPERATING EXPERIENCE + REACTOR, POWER + REACTOR, RESEARCH + REVIEW + FOREIGN EXCHANGE

135868
OSTERLE G
AN EXACT, STATIC, FINITE ELEMENT SOLUTION FOR TWO DIMENSIONAL STRESS BASED UPON THE THEORY OF THIN SHELLS (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-3 + GERRSR-185 +. 75 PPS, 26 FIGS, 7 REFS, NOV. 1977

135868 *CONTINUED*

STRUCTURE MECHANIC CALCULATIONS WHICH ARE BASED ON THE METHOD OF FINITE ELEMENTS, LEAD TO AN APPROXIMATIVE SOLUTION OF THE PROBLEM OF INTEREST IN NEARLY ALL CASES. NEVERTHELESS, SOME FEW ELEMENTS DELIVER EXACT STATIC SOLUTIONS, E.G., THE BAR AND THE BEAM ELEMENT INCLUDING BENDING, SHEARING AND TWISTING THEORY, SINCE THE ASSUMED DISPLACMENTS ARE IDENTICAL WITH THE SOLUTION OF THE DIFFERENTIAL EQUATION OF THE STATIC PROBLEM. THIS PAPER INTRODUCES ANOTHER FINITE ELEMENT, WHICH IS BASED ON THE THEORY OF THIN SHELLS AND YIELDS EXACT STATIC AND VERY ACCURATE DYNAMIC RESULTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STRESS ANALYSIS + SHELL + MATHEMATICAL TREATMENT + STRESS + FOREIGN EXCHANGE

136287
DESCRIPTIVE OF THE CURRENT PRACTICE REGARDING THE SAFETY CRITERIA FOR NUCLEAR POWER PLANTS. SAFETY CRITERIA 2.5 - DESIGN OF WORK AREAS, WORK CONDITIONS AND WORK ENVIRONMENT (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GERRSR-188 +. 85 PPS, TABS, NOV. 1977

NO ENGLISH ABSTRACT PROVIDED. THE DESIGN OF WORK AREAS, WORKING CONDITIONS, AND WORKING ENVIRONMENT ARE DISCUSSED.

GERMANY + *POWER PLANT, NUCLEAR + *SAFETY PROGRAM + *DESIGN CRITERIA + SAFETY ANALYSIS + STAFFING + ENVIRONMENT + SAFETY PRINCIPLES AND PHILOSOPHY + FOREIGN EXCHANGE

136288
SAFETY CRITERIA FOR NUCLEAR POWER PLANTS - RELEASE OF ATOMIC ENERGY WASTES INTO THE GROUND OF OCT. 12, 1977 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GERRSR-189 +. 21 PPS, OCT. 1977

NO ENGLISH ABSTRACT PROVIDED AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *POWER PLANT, NUCLEAR + *SAFETY PRINCIPLES AND PHILOSOPHY + SAFETY ANALYSIS + DESIGN CRITERIA + *WASTE DISPOSAL + WASTE DISPOSAL, TERRESTRIAL + FOREIGN EXCHANGE

136170
LERCHL G + LIESCH KJ
RESULTS OF LOFT EXPERIMENT L1-2 (CALCULATED WITH COMPUTER CODE BRUCH-D-06) (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-60 + GERRSR-192 +. 61 PPS, FIGS, 6 REFS, NOV. 1977

WITHIN THE FRAMEWORK OF THE AGREEMENT ON THE PROMOTION PROJECT RS 0182 UNDER THE SHORT TITLE "PARTICIPATION IN THE LOFT PROGRAM OF USNRC", THE GRS WAS ENGAGED IN POST TEST CALCULATIONS BASED ON THE INPUT DATA AS DELIVERED BY THE EXPERIMENTAL WORK. THE TEST L1-2 WAS RECALCULATED APPLYING THE COMPUTER CODE BRUCH-D-06. FOR THE RESULTS OF THESE COMPUTATIONS, COMPREHENSIVE COMPARISON MATERIAL GAINED FROM EXPERIMENTAL WORK WAS AVAILABLE. IT WAS THE AIM OF THESE INVESTIGATIONS, TO OPTIMALLY REPRESENT THE TEST PLANT, TO VERIFY THE INPUT DATA AND, IF NECESSARY, TO MODIFY THE PROGRAM IN ORDER TO SHOW THE THERMO- AND FLUIDDYNAMIC PHENOMENA IN THE PRIMARY SYSTEM OF A PWR DURING A LOCA, WITH CONSIDERATION TO ECONOMICAL ASPECTS (RELATIVE TO COMPUTER TIME).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + HYDRAULIC EXPERIMENT + PRESSURE TRANSIENT + THERMAL TRANSIENT + FOREIGN EXCHANGE + THERMAL EXPERIMENT

136291
BAIER G
CONTAINMENT RESEARCH PROGRAM RESULTS - BATTELLE INSTITUTE FRANKFURT ON COMPLETION OF CONTAINMENT RESEARCH TEST D6 OF RESEARCH PROJECT RS-50 - FINAL REPORT (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-64 + GERRSR-193 +. 41 PPS, 3 TABS, 27 FIGS, 5 REFS, NOV. 1977

PRESENTS PRE-TEST AND POST-TEST CALCULATIONS BY BATTELLE INSTITUTE, FRANKFURT UPON COMPLETION OF CONTAINMENT RESEARCH TEST D-6 OF RESEARCH PROJECT NUMBER RS-50. THIS IS THE FINAL REPORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *CONTAINMENT R AND D + MATHEMATICAL TREATMENT + FOREIGN EXCHANGE + CONTAINMENT ANALYSIS

136217
INVESTIGATIONS SPONSORED BY THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY ON RESEARCH PROJECTS IN THE AREA

136217 *CONTINUED*
OF REACTOR SAFETY. FOR THE PERIOD JULY 1-SEPTEMBER 30. 1977 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT. (GRS) MBH. F.R. GERMANY
GRS-F-67 + GERRSR-201 +. 413 PPS. TABS. FIGS. REFS. DEC. 1977

OBJECTIVE OF THIS PROGRAM IS TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FUTURE DEVELOPMENT OF SAFETY TECHNOLOGY. EACH PROGRESS REPORT REPRESENTS A COMPILATION OF INDIVIDUAL REPORTS ABOUT THE DIFFERENT PROJECTS OF THE RESEARCH PROGRAM. EACH REPORT CONTAINS AMONGST OTHER THINGS DETAILS ABOUT OBJECTIVES. THE WORK PERFORMED. THE RESULTS AND THE NEXT STEPS OF THE WORK.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*R AND D PROGRAM + *GERMANY + POWER PLANT. NUCLEAR + *SAFETY ANALYSIS + SAFETY MARGIN + SAFETY PROGRAM + FOREIGN EXCHANGE

136229
ACT ON THE PEACEFUL UTILIZATION OF ATOMIC ENERGY AND THE PROTECTION AGAINST ITS HAZARDS (ATOMIC ENERGY ACT OF OCTOBER 31. 1976)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH. F.R. GERMANY
GERRSR-202 +. 55 PPS. TRANSLATIONS-SAFETY CODES & GUIDES. EDITION 9/77. NOV. 1977

IT IS THE PURPOSE OF THIS ACT (1) TO PROMOTE RESEARCH. DEVELOPMENT AND UTILIZATION OF NUCLEAR ENERGY FOR PEACEFUL PURPOSES; (2) TO PROTECT LIFE. HEALTH AND PROPERTY AGAINST HAZARDS OF NUCLEAR ENERGY AND HARMFUL EFFECTS OF IONIZING RADIATION; AND TO PROVIDE COMPENSATION FOR DAMAGE CAUSED BY NUCLEAR ENERGY OR IONIZING RADIATION; (3) TO PREVENT DANGER TO THE INTERNAL OR EXTERNAL SECURITY OF THE FEDERAL REPUBLIC OF GERMANY ARISING FROM THE APPLICATION OR RELEASE OF NUCLEAR ENERGY; AND (4) TO ENABLE THE FEDERAL REPUBLIC OF GERMANY TO MEET ITS INTERNATIONAL OBLIGATIONS IN THE FIELD OF NUCLEAR ENERGY AND RADIOLOGICAL PROTECTION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

LEGISLATION + HAZARD. RELATIVE + INDUSTRY. NUCLEAR + ENERGY + RADIATION EFFECT + SECURITY + GERMANY + FOREIGN EXCHANGE

136230
ORDINANCE ON PROTECTION AGAINST DAMAGE AND INJURIES CAUSED BY IONIZING RADIATION (RADIOLOGICAL PROTECTION ORDINANCE OF OCTOBER 13. 1976)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH. F.R. GERMANY
GERRSR-203 +. 134 PPS. TRANSLATIONS-SAFETY CODES & GUIDES. EDITION 8/77. NOV. 1977

THIS ORDINANCE APPLIES TO THE HANDLING OF RADIOACTIVE SUBSTANCES (EXTRACTION. MANUFACTURE. STORAGE. TREATMENT. PROCESSING. OTHER USE AND DISPOSAL). TRADE IN RADIOACTIVE SUBSTANCES (ACQUISITION AND TRANSFER TO OTHERS). THE CARRIAGE. THE IMPORTING AND EXPORTING OF RADIOACTIVE SUBSTANCES AS WELL AS THE PROSPECTING. EXTRACTING AND PROCESSING OF RADIOACTIVE MINERALS. THIS ORDINANCE ALSO APPLIES TO THE CUSTODY OF NUCLEAR FUELS AND THE CONSTRUCTION AND OPERATION OF INSTALLATIONS FOR THE GENERATION OF IONIZING RADIATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

GERMANY + *RADIATION SAFETY AND CONTROL + FUEL HANDLING + WASTE HANDLING + MINING + ACCELERATOR + *REGULATION + SOURCE MATERIAL + BYPRODUCT MATERIAL + FOREIGN EXCHANGE + *TRANSPORTATION AND HANDLING

136231
COMPILATION OF INFORMATION ON CONTROL AND SHUTDOWN SYSTEMS REQUIRED FOR EXAMINATION PURPOSES IN LICENSING PROCEDURES OF NUCLEAR POWER PLANTS UNDER THE ATOMIC ENERGY ACT (STATE: NOVEMBER 30. 1976)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH. F.R. GERMANY
GERRSR-204 +. 17 PPS. TRANSLATIONS-SAFETY CODES & GUIDES. EDITION 3/78. 1977

THE PRESENT COMPILATION COMPRISES THE INFORMATION ON THE CONTROL AND SHUTDOWN SYSTEMS THAT IS REQUIRED FOR THE EXAMINATION OF LIGHT-WATER-COOLED REACTORS IN NUCLEAR POWER PLANTS. ONLY THOSE SYSTEMS ARE CONSIDERED WHOSE MAJOR TASK IS THE CONTROL AND SHUTDOWN OF THE REACTOR. I.E. THE CONTROL ASSEMBLIES AND THE COOLANT BORATION SYSTEMS. THE NECESSARY INFORMATION ON OTHER SYSTEMS WHICH DO NOT EXCLUSIVELY SERVE THE CONTROL OF THE REACTOR. SUCH AS THE COOLANT FLOW RATE CONTROL SYSTEM. WILL BE DEALT WITH IN SEPARATE COMPILATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

GERMANY + *LICENSING PROCESS + *PROCEDURES AND MANUALS + POWER PLANT. NUCLEAR + *CONTROL SYSTEM + *REACTOR SHUTDOWN + CONTROL RODS + REACTOR CONTROL + POISON. SOLUBLE + REACTOR. LWR + FOREIGN EXCHANGE

136232
COMPILATION OF INFORMATION ON REACTOR CORE REQUIRED FOR EXAMINATION PURPOSES IN LICENSING PROCEDURES OF NUCLEAR POWER PLANTS UNDER THE ATOMIC ENERGY ACT (STATE: NOVEMBER 30. 1976)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH. F.R. GERMANY
GERRSR-205 +. 19 PPS. TRANSLATIONS-SAFETY CODES & GUIDES. EDITION 2/78. 1977

136232 *CONTINUED*

THE PRESENT COMPILATION COMPRISES THE INFORMATION REQUIRED FOR EXAMINATION PURPOSES RELATING TO NUCLEAR CORE DESIGN, THERMOHYDRAULIC CORE DESIGN, FUEL AND FUEL RODS, CONTROL RODS AS TO THEIR REACTIVITY EFFECTIVENESS, BURNABLE POISONS AND BORON POISONING, AND CORE MONITORING INSTRUMENTATION AS TO ITS POSITIONING AND TASKS, OF LIGHT-WATER COOLED REACTORS IN NUCLEAR POWER PLANTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + POWER PLANT, NUCLEAR + LICENSING PROCESS + PROCEDURES AND MANUALS + REACTOR PHYSICS + THERMAL HYDRAULIC ANALYSIS + FUEL ROD + CONTROL RODS + REACTIVITY EFFECT + POISON, BURNABLE + POISON, SOLUBLE + REACTOR, LWR + INSTRUMENT, IN CORE + CORE + FOREIGN EXCHANGE

136233

COMPILATION OF INFORMATION ON THE EMERGENCY SUPPLY REQUIRED FOR EXAMINATION PURPOSES IN LICENSING PROCEDURES OF NUCLEAR POWER PLANTS UNDER THE ATOMIC ENERGY ACT (STATE: MARCH 18, 1976)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GERRSR-206 +. 8 PPS, TRANSLATION-SAFETY CODES & GUIDES, EDITION 1/78, 1977

THIS COMPILATION COMPRISES THE INFORMATION REQUIRED FOR THE EXAMINATION OF THE CONCEPT OF THE ENTIRE ELECTRIC ENERGY SUPPLY SYSTEM (MAINS CONNECTION, POWER PLANT CIRCUITRY, AUXILIARY POWER SUPPLY, CONNECTION OF THE ENERGY POWER SUPPLY SYSTEM) TO THE EXTENT IT IS RELEVANT FOR EMERGENCY POWER SUPPLY; THE CONCEPT OF THE EMERGENCY POWER SUPPLY SYSTEM; THE PARTS AND COMPONENTS OF THE EMERGENCY POWER SUPPLY SYSTEM, IN PARTICULAR GENERATING AND TRANSFORMING SYSTEMS AND THEIR AUXILIARY SYSTEMS, SWITCH AND DISTRIBUTION SYSTEM, CABLES AND LINES, SETUP LOCATIONS AND LAYING ROUTES, CONTROL, MONITORING AND REPORT FACILITIES; AND THE LIGHTNING PROTECTION AND GROUNDING FACILITIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *LICENSING PROCESS + *PROCEDURES AND MANUALS + POWER PLANT, NUCLEAR + ELECTRIC POWER + *EMERGENCY POWER, ELECTRIC + ELECTRIC POWER, AUXILIARY + TRANSFORMERS + CABLES AND CONNECTORS + LIGHTNING + PROTECTION SYSTEM + FOREIGN EXCHANGE

136180

BERNING A + MANSFELD G
EVALUATION OF THE MAIN EXPERIMENTS C6, C9, C13, AND C15 OF THE RESEARCH PROJECT RS 50 - PRESSURE DISTRIBUTION IN CONTAINMENT (BATTELLE-MODEL-CONTAINMENT) (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-71 + GERRSR-212 +. 168 PPS, TABS, FIGS, DEC. 1977

NO ENGLISH ABSTRACT WAS AVAILABLE WHEN THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + CONTAINMENT R AND D + TESTING + *CONTAINMENT + *PRESSURE, INTERNAL + *DISTRIBUTION + ANALYTICAL MODEL + FOREIGN EXCHANGE

136181

HELLINGS G
EVALUATION BY THE BATTELLE-INSTITUTE FRANKFURT/MAIN, PRESSURE CONDUCTOR TESTS D1 AND D3 OF RESEARCH PROJECT RS 50 "PRESSURE DISTRIBUTION IN CONTAINMENT" (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-72 + GERRSR-213 +. 83 PPS, FIGS, REFS, DEC. 1977

NO ENGLISH ABSTRACT WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + TESTING + SENSORS, PRESSURE + *PRESSURE, INTERNAL + *DISTRIBUTION + *CONTAINMENT + CONTAINMENT R AND D + FOREIGN EXCHANGE

137602

KTA SAFETY STANDARD 1501 - STATIONARY SYSTEM FOR MONITORING LOCAL DOSE RATES WITHIN NUCLEAR POWER PLANTS
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
EDITION 6/78 + GERRSR-222 +. 24 PPS, TABS, REFS, DEC. 15, 1977 (TRANSLATED JAN. 1978)

THIS SAFETY STANDARD SPECIFIES THE MEASUREMENT OBJECTIVE AND LAYS DOWN THE REQUIREMENTS FOR THE MEASUREMENT LOCATIONS, DESIGN WITH RESPECT TO THE MEASUREMENT OBJECTIVE, FAILURE AND HAZARD ALARMS, DISPLAY AND REGISTRATION OF MEASURED VALUES AND DOCUMENTATION, SPECIAL DESIGN FEATURES OF THE EQUIPMENT FOR MEASURING DURING AND AFTER INCIDENTS, MAINTENANCE AND REPAIR, AND TESTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*CODES AND STANDARDS + GERMANY + RADIATION MONITORS + MONITOR + *DOSE MEASUREMENT, EXTERNAL + DOSE + DESIGN CRITERIA + FOREIGN EXCHANGE

137745
KTA SAFETY STANDARD 2201.1 - DESIGN OF NUCLEAR POWER PLANTS AGAINST SEISMIC EVENTS; PART 1: BASIC PRINCIPLES
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
EDITION 5/78 + GERRSR-223 +. APPROX. 25 PPS, JULY 19, 1975 (TRANSLATED OCT. 1977)

THIS GUIDE DISCUSSES DESIGN EARTHQUAKES AND SAFETY EARTHQUAKES, AND IDENTIFIES EARTHQUAKE ZONES AND SEISMIC ENGINEERING DATA. PLANT COMPONENTS ARE CLASSIFIED, TYPES AND COMBINATIONS OF LOADS ARE DISCUSSED, AND CALCULATION TECHNIQUES ARE PRESENTED. OTHER CHAPTERS COVER STRESSES AND DEFORMATIONS, SEISMIC INSTRUMENTATION, AND SITING EFFECTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + CODES AND STANDARDS + POWER PLANT, NUCLEAR + SEISMOLOGY + EARTHQUAKE + EARTHQUAKE ENGINEERING + FOREIGN EXCHANGE

137603
KTA SAFETY STANDARD 2201.5 - DESIGN OF NUCLEAR POWER PLANTS AGAINST SEISMIC EVENTS; PART 5: SEISMIC INSTRUMENTATION
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
EDITION 6/78 + GERRSR-224 +. 8 PPS, AUG. 5, 1977 (TRANSLATED OCT. 1977)

THIS SAFETY STANDARD SPECIFIES THE REQUIREMENTS FOR A SEISMIC INSTRUMENTATION FOR THE VERIFICATION, ON ONE HAND WHETHER THE DESIGN PARAMETERS OF THE DESIGN EARTHQUAKE HAVE BEEN EXCEEDED, AND TO OBTAIN, ON THE OTHER HAND, INPUT PARAMETERS FOR AN ANALYTICAL VERIFICATION BY RECORDING EARTHQUAKE TIME HISTORIES. A SEISMIC INSTRUMENTATION IS SPECIFIED, WHOSE MEASURING PRINCIPLE IS BASED ON THE MEASUREMENT OF ACCELERATION, SINCE THIS CORRESPONDS TO CURRENT DESIGN PRACTICE AND INSTRUMENT TECHNOLOGY. A SEISMIC INSTRUMENTATION DIFFERENT FROM THE TYPE CHARACTERIZED HERE CAN BE USED IF IT ALLOWS TO DETERMINE THE PARAMETERS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CODES AND STANDARDS + SEISMIC DESIGN + MONITOR + INSTRUMENT, SURVEILLANCE + INSTRUMENTS, MISC. + MEASUREMENT + *EQUIPMENT DESIGN + *INSTRUMENT, EARTHQUAKE + GERMANY + FOREIGN EXCHANGE

137604
AIR LOCKS THROUGH THE CONTAINMENT VESSEL OF NUCLEAR POWER PLANTS - PERSONNEL LOCKS - KTA SAFETY STANDARD 3402
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
EDITION 7/78 + GERRSR-225 +. 18 PPS, 1 TAB, FEB. 24, 1977 (TRANSLATED OCT. 1977)

THIS SAFETY STANDARD APPLIES TO PERSONNEL LOCKS THROUGH THE CONTAINMENT VESSEL OF NUCLEAR POWER PLANTS. THIS SAFETY STANDARD DOES NOT APPLY TO AIR LOCKS WHICH ARE INTENDED EXCLUSIVELY FOR TRANSFERRING MATERIALS AND WHICH ARE NOT PERMITTED TO BE USED FOR TRANSFERRING PERSONS. PERSONNEL LOCKS HAVE THE FOLLOWING SAFETY FUNCTIONS: PREVENTION OF INADMISSIBLE RELEASES OF RADIOACTIVE MATERIALS INTO THE ENVIRONMENT. THIS ALSO APPLIES DURING TRANSFER OPERATIONS THROUGH THE AIR LOCK. TO MAKE POSSIBLE ESCAPE AND RESCUE OPERATIONS FROM THE REACTOR CONTAINMENT VESSEL. PREVENTION OF UNCONTROLLED ACCESS TO THE CONTAINMENT VESSEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIRLD, VA. 22161

GERMANY + *CODES AND STANDARDS + *CONTAINMENT AIR LOCK + POWER PLANT, NUCLEAR + CONTAINMENT INTEGRITY + DESIGN CRITERIA + REGULATION, FEDERAL + FOREIGN EXCHANGE

137605
KTA SAFETY STANDARD 3501 - REACTOR PROTECTION SYSTEM AND MONITORING OF ENGINEERED SAFEGUARDS
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
EDITION 8/78 + GERRSR-226 +. 59 PPS, JUNE 11, 1977 (TRANSLATED OCT. 1977)

THIS SAFETY STANDARD CONTAINS DESIGN CRITERIA AND REQUIREMENTS FOR THE RELIABILITY AND FUNCTIONAL PERFORMANCE OF THE REACTOR PROTECTION SYSTEM. IT COVERS THE DESIGN, CONSTRUCTION, EQUIPMENT QUALITY, INSTALLATION AND TESTING OF THE REACTOR PROTECTION SYSTEM AND ITS COMPONENTS. REQUIREMENTS NOT SPECIFICALLY RELATED TO REACTOR PROTECTION OR ALREADY LAID DOWN IN ENGINEERING STANDARDS ARE NOT DEALT WITH HEREIN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *CODES AND STANDARDS + *REACTOR PROTECTION SYSTEM + ENGINEERED SAFETY FEATURE + DESIGN CRITERIA + FOREIGN EXCHANGE

137601
KTA SAFETY STANDARD 3902 - LIFTING EQUIPMENT IN NUCLEAR FACILITIES
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
EDITION 9/78 + GERRSR-227 +. 31 PPS, FEB. 3, 1976 (TRANSLATED OCT. 1977)

THIS SAFETY STANDARD APPLIES TO ELEVATORS, CRANES, WINCHES, TELPHERS, LOAD-BEARING EQUIPMENT AND REFUELING MACHINES FOR LIGHT WATER REACTORS - IN THE FOLLOWING COLLECTIVELY CALLED LIFTING

137601 *CONTINUED*
EQUIPMENT - IN AS FAR AS THEY ARE USED IN FACILITIES FOR PRODUCING OR FISSIONING NUCLEAR FUEL OR FOR REPROCESSING IRRADIATED NUCLEAR FUEL OR FOR STORING OR OTHER USES OF NUCLEAR FUEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + *CODES AND STANDARDS + *CRANE + *FUEL HANDLING MACHINE + EQUIPMENT + REFUELING + FUEL HANDLING + FOREIGN EXCHANGE

137606
LIST OF REPORTS ON REACTOR SAFETY RESEARCH BY BMFT, USNRC, EPRI AND USTA, REPORTING PERIOD OCTOBER 1-DECEMBER 31, 1977 (IN ENGLISH & GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-F-50 + GERRSR-229 +. 41 PPS, FEB. 1978

THIS LIST REVIEWS REPORTS FROM THE FEDERAL REPUBLIC OF GERMANY, FROM THE UNITED STATES OF AMERICA AND FROM JAPAN CONCERNING SPECIAL PROBLEMS IN THE FIELD OF REACTOR SAFETY RESEARCH. THE LIST PURSUES THE FOLLOWING ORDER: COUNTRY OF ORIGIN, PROBLEM AREA CONCERNED, ACCORDING TO THE REACTOR SAFETY RESEARCH PROGRAM OF BMFT, REPORTING ORGANIZATION. THE LIST OF REPORTS APPEARS QUARTERLY. THE REPORT PERIOD COVERS OCTOBER-DECEMBER 1977.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*REVIEW + REACTOR + *R AND D PROGRAM + SAFETY PROGRAM + GERMANY + UNITED STATES + JAPAN + EPRI + AGENCY, NRC + BIBLIOGRAPHY + FOREIGN EXCHANGE

137464
HOLM D
THEORETICAL CALCULATION OF THE COMPOSITION OF NOBLE GAS ACTIVITY IN THE GASEOUS EFFLUENTS OF NUCLEAR POWER PLANT WITH PRESSURIZED WATER REACTORS (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-8 + GERRSR-230 +. 24 PPS, 4 TABS, 1 FIG, 5 REFS, JAN. 1978

SHOWS A POSSIBLE MEANS OF CALCULATING THEORETICALLY THE COMPOSITION OF NOBLE GAS ACTIVITY IN GASEOUS EFFLUENTS OF NUCLEAR POWER PLANTS WITH PRESSURIZED WATER REACTORS. THE CALCULATIONS OF THE NOBLE GAS NUCLIDE SPECTRA ARE BASED ON GENERATED DATA OF THE ACTIVITY SOURCE TERMS IN THE REACTOR CORE AND PLANT-SPECIFIC DESIGN FEATURES. HOW THE NOBLE GAS NUCLIDE SPECTRUM VARIES BY ALTERING THE DISCHARGE PATHWAY COMBINATION IS EXAMINED. THE PROPOSED MATHEMATICAL MODEL WAS VERIFIED VIA REAL MEASUREMENT DATA. FOR A NOBLE GAS ACTIVITY DISCHARGE OF 40,000 CI PER YEAR FROM PWRS THIS TIME BY KRAFTWERK UNION AG (KWU) A SUFFICIENT CONSERVATIVE NUCLIDE SPECTRUM IS GIVEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FISSION PRODUCT RELEASE + FISSION GAS RELEASE + ANALYTICAL MODEL + ENERGY SPECTRUM + THEORETICAL INVESTIGATION + FOREIGN EXCHANGE + NOBLE GAS

132482
RICHTER G + HENNERT G
CALCULATION OF RELIABILITY DATA FOR COMPLEX SYSTEMS IN USING ANALYTICAL SUCCESS PATHS METHODS (IN GERMAN)
INSTITUT FUR KERNTECHNIK, GERMANY
TUBIK-28 + BMFT-RS 106 + GERRSR-168 +. 57 PPS, FIGS, REFS, OCT. 1973

COMPUTER PROGRAMS WERE DEVELOPED TO CALCULATE THE RELIABILITY DATA OF REDUNDANT COMPLEX SYSTEMS WITH ANALYTICAL METHODS. THE SYSTEMS ARE REPRESENTED BY LOGICAL BLOCK DIAGRAMS WHERE ALL SUCCESS PATHS ARE EVALUATED BY THE COMPUTER PROGRAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

RELIABILITY ANALYSIS + *RELIABILITY, SYSTEM + ANALYTICAL TECHNIQUE + *COMPUTER PROGRAM + FOREIGN EXCHANGE

132481
RICHTER G
CALCULATION OF THE RELIABILITY OF LARGE COMPLEX SYSTEMS USING THE RELEVANT PATH METHOD
INSTITUT FUR KERNTECHNIK, GERMANY
TUBIK-46 + BMFT RS 106 + GERRSR-169 +. 150 PPS, FIGS, REFS, MARCH 1975

QUANTIFIED RELIABILITY EXAMINATION OF COMPLEX SYSTEMS HAS BEEN USED WIDELY IN RECENT YEARS, MOSTLY TO EVALUATE THE SAFETY OF THE SYSTEMS; HOWEVER, ECONOMIC QUESTIONS MUST BE ALSO TAKEN INTO ACCOUNT. QUANTITATIVE ANALYSES HELP TO EVALUATE THESE SYSTEMS WITH THEIR HIGH REQUIREMENTS ON SAFETY BECAUSE THIS LIMITS THE RISKS. THE PROBABILISTIC DETERMINATION OF THE SAFETY FACTORS OF THE SYSTEM REQUIRES USUALLY A LARGE COMPUTER PROGRAM. FOR LARGE SYSTEMS THE LESS EXPENSIVE DEDUCTIVE FAULT-TREE AND BLOCK DIAGRAM METHOD IS USED. AN ANALYTICAL METHOD WAS DEVELOPED AND TESTED TO DETERMINE THE RELIABILITY FACTORS OF ENGINEERING SYSTEMS IN CASE OF GIVEN FAULT-TREE SITUATIONS.

132481 *CONTINUED*
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

RELIABILITY ANALYSIS + ECONOMICS + PROBABILITY + *RELIABILITY, SYSTEM + COMPARISON + FAULT TREE ANALYSIS + CONTROL SYSTEM + *ANALYTICAL TECHNIQUE + FOREIGN EXCHANGE

135312
JAHN H
HOLOGRAPHIC INVESTIGATION OF THE FREE CONVECTION IN A CORE MELTDOWN (IN GERMAN)
INSTITUT FUR VERFAHRENSTECHNIK, F.R. GERMANY
BMFT RS 48/1 + GERRSR-161 +. 156 PPS. TABS. FIGS. 146 REFS. 1975

CORE MELTDOWN HAS BEEN STUDIED INTENSIVELY. CONSIDERING FOR EXAMPLE THE THEORETICAL CORE MELTDOWN ACCIDENT OF A SOLID FUEL REACTOR WHICH REPRESENTS A LIQUID SYSTEM HEATED BY AFTERHEAT WITH EXTERNAL COOLING, RESULTING IN NATURAL CONVECTION. DESIGN PROBLEMS OF HOMOGENEOUS REACTORS ARE ALSO CONSIDERED; THE HEAT GENERATED BY THE NUCLEAR REACTION IN THE LIQUID FUEL IS REMOVED BY FREE CONVECTION. AN EXPERIMENTAL METHOD WAS USED TO STUDY THE HEAT TRANSFER FROM A LIQUID WITH INTERNAL HEAT SOURCES TO THE HEAT CONDUCTING WALLS BY MEANS OF NATURAL CONVECTION IN A CLOSED CAVITY. THE CONVECTION OF THE LIQUID WAS OBSERVED BY MEANS OF INTERFEROMETRIC HOLOGRAPHY. FOR THE CONTROL EXPERIMENTS BENARD CONVECTION PHENOMENA WERE USED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR + *ACCIDENT, HYPOTHETICAL + *CORE MELTDOWN + EXPERIMENT + ACCIDENT ANALYSIS + FOREIGN EXCHANGE

135313
REINEKE HH + SCHRAMM R + STEINBERNER U
BEHAVIOR OF THE CORE MELTDOWN IN CASE OF A THEORETICAL ACCIDENT. ANNUAL REPORT FOR 1975 (IN GERMAN)
INSTITUT FUR VERFAHRENSTECHNIK, F.R. GERMANY
BMFT RS 166 + GERRSR-166 +. 82 PPS. TABS. FIGS. MARCH 1976

PRESENTS A SUMMARY OF RESEARCH WORK DONE IN 1975 ON PROJECT RS166 CONCERNING BEHAVIOR OF THE MOLTEN CORE IN A HYPOTHETICAL REACTOR ACCIDENT. THE FIRST PART DESCRIBES THE INVESTIGATION OF THE THERMAL-HYDRAULIC BEHAVIOR AND HEAT TRANSFER IN A POOL OF MOLTEN CORE MATERIAL AT THE BOTTOM OF THE PRESSURE VESSEL. THE INVESTIGATIONS ARE DONE BOTH NUMERICALLY (USING FINITE DIFFERENCE METHODS) AND EXPERIMENTALLY (USING HOLOGRAPHIC INTERFEROMETRY AND A LASER-DOPPLER-ANEMOMETER). THE SECOND PART OF THE REPORT EXPLAINS THE INVESTIGATION OF THE THERMAL-HYDRAULIC INTERACTION BETWEEN MOLTEN CORE MATERIAL AND MOLTEN CONCRETE. A COMPUTER CODE WAS DEVELOPED WHICH IS ABLE TO DESCRIBE THE FREE CONVECTION IN A SYSTEM OF TWO LIQUID PHASES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*EXPERIMENT + COMPUTER PROGRAM + *ANALYTICAL MODEL + ACCIDENT ANALYSIS + REACTOR, LWR + *ACCIDENT, HYPOTHETICAL + THERMAL HYDRAULIC ANALYSIS + *CORE MELTDOWN + FOREIGN EXCHANGE

132512
MOHR W + REPPLINGER W
ELECTRODYNAMIC CONTACT-FREE EXCITATION OF FREE ULTRASONIC WAVES. 3. ENGINEERING AND SCIENTIFIC REPORT CONCERNING PROJECT RS 102-18 PART II "ELECTRODYNAMIC METHODS" (IN GERMAN)
INSTITUT FUR ZERSTORUNGSFREIE PRUFVERFAHREN, GERMANY
BMFT RS 102-18, PART 2 + GERRSR-162 +. 69 PPS. FIGS. REFS. MARCH 30, 1977

THE EFFICIENCY RATINGS OF ELECTRODYNAMIC AND PIEZOELECTRIC TRANSDUCERS WERE COMPARED; THE FORMER EXHIBITED A GREAT POTENTIAL FOR FURTHER DEVELOPMENT. THESE ARE SEVERAL PROBABILITIES IN THE FIELD OF REACTOR SAFETY FOR CONTACT-FREE ULTRASONIC TESTING WITH THIS INSTRUMENT, SUCH AS HASTING OF THE SNR REACTOR CONTAINER, EXTERNAL TESTING OF LIGHT WATER REACTORS, TESTING OF COMPONENTS DURING FINISHING. THE CONTACT-FREE ELECTRODYNAMIC EXCITATION OF ULTRASOUND HAS BEEN SUCCESSFULLY CARRIED OUT IN GUIDED WAVES. IMPROVED ELECTRIC MATCHING, MINIMIZING THE SKIN AND PROXIMITY EFFECTS AND THE USE OF PULSE INSTEAD OF STATIC MAGNETIZING HELPED TO IMPROVE H. LICHT'S PLATE WAVE TRANSDUCER SO THAT IT CAN BE USED AS A TESTING HEAD FOR FREE ULTRASONIC WAVES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ULTRASONICS + *EQUIPMENT DEVELOPMENT + INSTRUMENT, COMPONENT + INSTRUMENT, SURVEILLANCE + FOREIGN EXCHANGE

135324
KRAUS S + GOEBBELS K
A SIGNAL AVERAGING METHOD TO SUPPRESS THE COHERENT BACKGROUND DURING THE FAULT TESTING OF LARGE-GRAIN STRUCTURAL MATERIALS. FOURTH INTERIM REPORT ON PROJECT RS 102-16 (IN GERMAN)
INSTITUT FUR ZESTORUNGSFREIE PRUFVERFAHREN, GERMANY
BMFT RS 102-16 + GERRSR-165 +. 55 PPS. 10 FIGS. 44 REFS. FEB. 2, 1977

ULTRASONIC TESTING OF STEELS IS USED FOR THE NONDESTRUCTIVE TESTING OF REACTOR COMPONENTS. AUSTENITIC AND PLATED MATERIALS PRESENT PROBLEMS: THE ULTRASOUND IS MORE STRONGLY ALTERNATED THAN FERRITIC STEELS AND THE STRONG SCATTERING REFLECTIONS RENDER THE RECOGNITION OF FAULTS MORE DIFFICULT, BECAUSE THEIR AMPLITUDE MAY COVER THE USEFUL SIGNALS. VARIOUS SIGNAL-AVERAGING

135324 *CONTINUED*

METHODS WERE DEVELOPED TO IMPROVE THE SIGNAL/NOISE RATIO. THE METHODS WERE DESIGNATED AS PLACE AVERAGING, FREQUENCY AVERAGING AND ANGLE AVERAGING. THE MOST IMPORTANT RESULT OF THIS STUDY IS THE REPRESENTATION OF THE QUANTITATIVE EVALUATION OF THE AVERAGING CURVES ACCORDING TO THE AVG METHOD, USING INFORMATION-HANDLING METHODS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ULTRASONICS + TEST, NONDESTRUCTIVE + FAULT + POWER PLANT, NUCLEAR + COMPONENTS + FAILURE, INSTRUMENT + FOREIGN EXCHANGE

137456
MURAO Y
AN ANALYTICAL STUDY OF THE THERMO-HYDRODYNAMIC BEHAVIOR OF THE REFLOOD-PHASE DURING A LOCA (IN ENGLISH)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2545 + GERRSR-194 +. 83 PPS, TABS, FIGS, 21 REFS, DEC, 1977

THE OBJECTIVES OF THIS STUDY ARE: THE CHECK OF THE QUENCH MODEL PROPOSED BY THE AUTHOR AND T. SUDOH, THE ESTABLISHMENT OF THE THERMO-HYDRODYNAMICS DOWNSTREAM FROM THE QUENCH FRONT, AND THE STABILIZATION OF THE NUMERICAL CALCULATIONS. IN ORDER TO STUDY THESE TERMS, THE NEW VERSION OF THE REFLOOD ANALYSIS CODE "REFLA-1D" WAS DEVELOPED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ANALYTICAL MODEL + *ACCIDENT, LOSS OF COOLANT + *THERMAL HYDRAULIC ANALYSIS + *CORE REFLOODING + COMPUTER PROGRAM + COMPARISON, THEORY AND EXPERIENCE + REACTOR, LWR + FOREIGN EXCHANGE

136549
PENZHORN R-D
ALTERNATE PROCESSES FOR THE LONG TERM STORAGE OF KR-85 (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2482 + GERRSR-195 +. 67 PPS, TABS, FIGS, 91 REFS, DEC, 1977

THE FORSEEABLE PROBLEMS WHICH ARE CHARACTERISTICS OF THE LONG-TERM STORAGE OF KR-85, ARE DISCUSSED ON THE BASIS OF THE CHEMICAL, PHYSICAL AND RADIOLOGICAL PROPERTIES OF THE FISSION NOBLE GASES KRYPTON AND XENON AS WELL AS RUBIDIUM, THE DAUGHTER NUCLIDE OF KR-85. THE VARIOUS ALTERNATIVES SUGGESTED IN THE LITERATURE FOR THE LONG-TERM STORAGE OF KR-85 ARE REVIEWED. THE TECHNOLOGICAL REALIZATION OF EACH OF THE PROPOSED ALTERNATIVES IS ANALYZED.

AVAILABILITY - INIS SECTION, INTERNATIONAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA

FISSION GAS RELEASE + WASTE STORAGE + PROPERTY, CHEMICAL + PROPERTY, PHYSICAL + KRYPTON + XENON + RUBIDIUM + FOREIGN EXCHANGE

136171
CLASS G
THEORETICAL INVESTIGATION OF PRESSURE PULSATIONS OCCURRING DURING STEAM CONDENSATION IN THE BWR-PRESSURE SUPPRESSION SYSTEM - COMPUTER CODE KONDAS (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2487 + GERRSR-196 +. 152 PPS, 41 FIGS, 19 REFS, OCT, 1977

DURING CONDENSATION OF STEAM IN THE BWR PRESSURE SUPPRESSION SYSTEM PRESSURE PULSATIONS ARE GENERATED. MODEL APPROACHES ARE ESTABLISHED AND DISCUSSED WHICH PERMIT A THEORETICAL DESCRIPTION OF THE PRINCIPAL PROCESSES. IT APPEARS THAT SOME OF THE BASIC PHENOMENA HAVE STILL TO BE TREATED BY RELATIVELY COARSE ASSUMPTIONS AND SIMPLIFICATIONS. THIS HOLDS IN PARTICULAR FOR THE BEHAVIOR OF LIQUID SURFACE AND OF STEAM BUBBLES, AND THE HEAT AND MASS TRANSFER ON FREE SURFACE OF FLUIDS. IN THE RECENTLY ESTABLISHED KONDAS COMPUTER PROGRAM CORRECTION PARAMETERS ARE USED FOR UNCERTAIN ASSUMPTIONS. IT IS SHOWN HOW THE COMPUTER PROGRAM COULD BE ADAPTED AND VERIFIED. THE DEPENDENCIES OF THE PRESSURE PULSATIONS ON VARIOUS INFLUENCES ARE INDICATED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + *CONDENSATION + REACTOR, BWR + STEAM + PRESSURE PULSE + PRESSURE TRANSIENT + *CONTAINMENT, PRESSURE SUPPRESSION + FOREIGN EXCHANGE

136172
WIEHR K + SCHMIDT HE
OUT-OF-PILE EXPERIMENTS ON BALLOONING OF ZIRCALOY FUEL ROD CLADDINGS. TEST RESULTS WITH SHORTENED FUEL ROD SIMULATORS
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2395 + GERRSR-197 +. 28 PPS, 13 FIGS, 4 REFS, OCT, 1977

BALLOONING EXPERIMENTS WERE PERFORMED ON SHORTENED INDIRECTLY HEATED FUEL FOR SIMULATORS IN AN AIR ATMOSPHERE. THE TEST SETUP AND THE TEST PROCEDURE ARE DESCRIBED. THE FOLLOWING RESULTS WERE ACHIEVED: INITIAL INTERNAL ROD PRESSURES OF 70 BAR RESULT IN BURST CIRCUMFERENTIAL STRAINS OF ABOUT 38 %. INITIAL ECCENTRICITIES BETWEEN THE PELLET STACK AND THE CLADDING INITIATE

136172 *CONTINUED*

DIFFERENTIAL CIRCUMFERENTIAL TEMPERATURES UP TO 100 K AT RUPTURE. IN CASE OF PRONOUNCED DIFFERENTIAL CIRCUMFERENTIAL TEMPERATURES THE RUPTURE POINT IS ALWAYS LOCATED ON THE HOT SURFACE LINE WHICH, IN MOST CASES, IS A STRAIGHT OR PARTLY CONCAVE LINE WITH HIGH LOCAL STRAIN, I.E., MAXIMUM WEAKENING OF THE WALL THICKNESS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ZIRCALOY + FUEL ROD + CLADDING + SIMULATION + SWELLING + FOREIGN EXCHANGE

136550
AMMON RV + STRAUCH K + WURSTER W
THE CATALYSTIC REDUCTION OF OXYGEN AND NITRIC OXIDES WITH HYDROGEN IN THE OFFGAS OF REPROCESSING PLANTS
KERFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2437 + GERRSR-198 +. 39 PPS, 11 TABS, 6 FIGS, 28 REFS, MAY 1977

THE ACTIVITY AND SELECTIVITY OF 13 COMMERCIAL NOBLE METAL CATALYSTS TOWARDS THE REDUCTION OF O2 AND NO WITH H2 WERE TESTED. THE BEHAVIOR OF SEVERAL SELECTED CATALYSTS TOWARDS POISONING BY ORGANIC PHOSPHATE (TRIBUTYLPHOSPHATE) AND IODINE AS WELL AS TOWARDS EXCESSIVE THERMAL TREATMENT WAS ALSO STUDIED. THE RESULTS INDICATE THAT RU CATALYSTS HAVE SUPERIOR PROPERTIES OVER PD AND PT CATALYSTS FOR AN APPLICATION IN THE OFFGAS PURIFICATION SYSTEM OF REPROCESSING PLANTS.

AVAILABILITY - INIS SECTION, INTERNATICNAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA

FUEL REPROCESSING + REDUCTION + CATALYST + CHEMICAL REACTION + OXIDATION + OFF GAS + IODINE + OXYGEN + NITROGEN + RUTHENIUM + HYDROGEN + PLATINUM + FOREIGN EXCHANGE + PALLADIUM

136176
ALSMEYER H + BARLEON L + KOSTER J
A MODEL DESCRIBING THE INTERACTION OF A CORE MELT WITH CONCRETE (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2395 + GERRSR-199 +. 49 PPS, 7 TABS, 12 FIGS, 20 REFS, OCT. 1977

FOR THIS PROCESS A THEORETICAL MODEL IS DERIVED DESCRIBING THE DECOMPOSITION RATE, WHICH DEPENDS ON THE TEMPERATURE OF THE CORE MELT. THE MODEL IS VALID UNDER THE ASSUMPTIONS THAT THE VOLUME FLUX OF THE GASEOUS DECOMPOSITION PRODUCTS IS SUFFICIENT TO MAINTAIN A CONTINUOUS GAS LAYER BETWEEN MELT AND CONCRETE AND THAT NO CRUSTS ARE FORMED BETWEEN MELT AND CONCRETE. FROM THE GAS FILM, BUBBLES ARE DETACHED AND FLOW THROUGH THE CORE MELT. FOR THE CASE OF A HOMOGENOUS METALLIC PHASE A DIFFUSION MODEL IS FORMULATED FOR THE OXIDATION OF STEEL AND FOR THE REDUCTION OF STEAM TO HYDROGEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONCRETE + CORE MELTDOWN + DECOMPOSITION + GAS + FILM + TESTING + BUBBLE + ANALYTICAL MODEL + FOREIGN EXCHANGE

136228
BARSCHDORFF D + NEUMANN M + WIOGO S
DATA EVALUATION OF INFRARED ABSORPTION MEASUREMENTS AT BLOWDOWN EXPERIMENTS WITHIN THE DEPRESSURISATION SYSTEM OF THE MARVIKEN REACTOR (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY

AN INFRARED MEASURING INSTRUMENTATION FOR THE DETERMINATION OF THE PARAMETERS OF TRANSIENT TWO PHASE/MULTICOMPONENT FLOW IS DEVELOPED. IT WAS OPERATED FOR THE FIRST TIME IN LARGE SCALE EXPERIMENTS AT THE MARVIKEN MX II-BLOWDOWN SERIES. AFTER A SHORT DESCRIPTION OF THE INSTRUMENTATION, IN THIS REPORT THE EARLIER PUBLISHED UNDERLYING THEORETICAL MODEL IS FURTHER DEVELOPED AND THE DATA EVALUATION IS TREATED IN DETAIL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*EQUIPMENT DEVELOPMENT + INSTRUMENT, TEMPERATURE + *INSTRUMENT, FLOW + *FLOW, TWO PHASE + COMPONENT DESCRIPTION + EQUIPMENT DESIGN + MEASUREMENT + BLOWDOWN + FOREIGN EXCHANGE + MARVIKEN (BHWR)

137597
197/ SEMIANNUAL REPORT - PROJECT NUCLEAR SAFETY (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2500 + GERRSR-228 +. 573 PPS, FIGS, REFS, DEC. 1977

THE 11TH PROJECT REPORT 1/77 IS A DESCRIPTION OF WORK WITHIN THE NUCLEAR SAFETY PROJECT PERFORMED IN THE FIRST SIX MONTHS OF 1977 IN THE NUCLEAR SAFETY FIELD BY GFK INSTITUTES AND DEPARTMENTS AND BY EXTERNAL INSTITUTIONS ON BEHALF OF GFK. IT INCLUDES FOR EACH INDIVIDUAL RESEARCH ACTIVITY SHORT SUMMARIES (PP. 52-113) IN ENGLISH ON WORK COMPLETED, ESSENTIAL RESULTS, AND PLANS FOR THE NEAR FUTURE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*GERMANY + *R AND D PROGRAM + SAFETY ANALYSIS + REACTOR TRANSIENT + ENVIRONMENTAL QUALITY + POWER PLANT.

137597 *CONTINUED*
NUCLEAR + FUEL REPROCESSING + ZIRCALOY + ACCIDENT ANALYSIS + FOREIGN EXCHANGE + FUEL ROD.

132849
KNODLER D
BEHAVIOR OF ZIRCALOY CLADDING UNDER STRAIN ARISING IN LOSS-OF-COOLANT ACCIDENTS
KRAFTWERK UNION, ERLANGEN, F.R., GERMANY
BMFT RS 107 + GERRSR-152 +. 128 PPS, 3 TABS, 56 FIGS, 41 REFS, MARCH 1977

AN EXPERIMENTAL PARAMETER STUDY WAS PERFORMED TO INVESTIGATE THE STRAIN AND RUPTURE BEHAVIOR OF ZIRCALOY-4 CLADDING SAMPLES UNDER SIMULATED LOCA-CONDITIONS. PARAMETER RANGES AND COMBINATIONS WERE EVALUATED IN WHICH THE CLADDING EXPERIENCES A REMARKABLE DEFORMATION. SCOPING TESTS WITH DIFFERENT TEST TECHNIQUES AND UNDER DIFFERENT TEST CONDITIONS HAVE SHOWN THAT THE RESULTS OF THE PARAMETER STUDY WITH DIRECTLY HEATED SPECIMENS IN AIR ATMOSPHERE DESCRIBE THE MAXIMUM DUCTILITY OF THE CLADDING AND ARE OVER-CONSERVATIVE IN RESPECT TO THE BEHAVIOR OF A FUEL ROD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*ACCIDENT, LOSS OF COOLANT + EXPERIMENT + *ZIRCALOY + STRESS RUPTURE + *FAILURE, CLADDING + FOREIGN EXCHANGE + STRESS STRAIN DATA

135841
MARKL H
MODEL CALCULATIONS TO VERIFY THE SHORT-TERM BEHAVIOR OF ABSOLUTE AND DIFFERENTIAL PRESSURES MEASURED IN THE RS 50 EXPERIMENTS C13 AND C15 (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R., GERMANY
RS 50-24-4 + GERRSR-178 +. 115 PPS, TABS, FIGS, 11 REFS, SEPT. 1977

THE PRESENT REPORT COVERS THE CONTAINMENT EXPERIMENTS RS 50, SERIES C. THE ABSOLUTE AND DIFFERENTIAL PRESSURES OCCURRING IN THE INDIVIDUAL COMPARTMENTS UPON RUPTURE OF THE COOLANT PIPE ARE CALCULATED AND COMPARED WITH THE MEASURED DATA. ALL CALCULATIONS WERE CARRIED OUT BY MEANS OF THE DDIFF COMPUTER PROGRAM. THE VALUES SELECTED FOR THE INPUT PARAMETERS CONTRACTION COEFFICIENT ALPHA AND WATER ENTRAINMENT FACTOR C CORRESPOND TO THOSE USED IN THE DESIGN CALCULATIONS FOR REAL POWER PLANTS. THE CALCULATIONS HAVE BEEN FOUND TO YIELD CONSERVATIVE VALUES OF ABSOLUTE AND DIFFERENTIAL PRESSURES IN PARTICULAR FOR THE RUPTURE COMPARTMENT AND THE IMMEDIATELY FOLLOWING COMPARTMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTAINMENT + EXPERIMENT + COMPUTER PROGRAM + COMPARTMENT + COMPARISON, THEORY AND EXPERIENCE + COMPARISON + FOREIGN EXCHANGE + PRESSURE TRANSIENT

135844
MARKL H
MODEL CALCULATIONS TO VERIFY THE SHORT-TERM BEHAVIOR OF ABSOLUTE AND DIFFERENTIAL PRESSURES MEASURED IN THE RS 50 EXPERIMENTS C13 AND C15 (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R., GERMANY
RS 50-24-5 + GERRSR-179 +. 70 PPS, FIGS, REFS, SEPT. 1977

THE PRESENT REPORT COVERS THE CONTAINMENT EXPERIMENTS RS 50, SERIES C, NOS. C13 AND C15. THE PRESSURE HISTORIES FOR THE FIRST 2.0 SECONDS AND THE RESULTANT DIFFERENTIAL PRESSURES ARE THEORETICALLY DETERMINED AND COMPARED WITH THE MEASURED VALUES. THE CALCULATIONS WERE CARRIED OUT BY MEANS OF THE COMPUTER PROGRAM DDIFF. THE SHAPES OF ALL CALCULATED ABSOLUTE PRESSURE PLOTS ARE SIMILAR TO THOSE OF THE MEASURED PRESSURE PLOTS. THE INDIVIDUAL VALUES OF THE CALCULATED ABSOLUTE PRESSURE, HOWEVER, ARE ALWAYS FAIRLY TO HIGHLY CONSERVATIVE - IN PARTICULAR FOR THE RUPTURE COMPARTMENT AND THE IMMEDIATELY FOLLOWING COMPARTMENTS. THE MOST CONSERVATIVE VALUES WERE OBTAINED IN THOSE CALCULATIONS WHICH CORRESPOND TO THE DESIGN CALCULATIONS FOR REAL POWER PLANTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTAINMENT + EXPERIMENT + COMPUTER PROGRAM + COMPARTMENT + COMPARISON, THEORY AND EXPERIENCE + COMPARISON + FOREIGN EXCHANGE + PRESSURE TRANSIENT

136594
HERING W + MANZEL R
MEASURED FISSION GAS RELEASE DATA FROM PWR FUEL RODS
KRAFTWERK UNION, ERLANGEN, F.R., GERMANY
GERRSR-210 +. 4 PPS, 1 TAB, 1977

THIS REPORT DESCRIBES THE RESULTS OF FISSION GAS RELEASE MEASUREMENTS PERFORMED BY KRAFTWERK UNION. THE RESULTS PRESENTED HAVE BEEN OBTAINED FROM THE OBRIGHEIM PRESSURIZED WATER REACTOR AND REFER TO STANDARD FUEL RODS, HIGH-POWER EXPERIMENTAL RODS AND FUEL RODS FROM A CYCLING EXPERIMENT. ALL FUEL RODS HAVE ZIRCALOY CLADDINGS. THE FUEL RODS WERE PIERCED AT A POINT WITHIN THE UPPER GAS PLENUM INSIDE AN EVACUATED CONTAINER. THE OVERFLOWING GAS WAS PUMPED OUT AND THE QUANTITY OF GAS AT N.T.P. WAS CALCULATED FROM THE SYSTEM PRESSURE RISE. SUBSEQUENTLY A GAS

136594 *CONTINUED*
SAMPLE WAS TAKEN FOR MASS SPECTROMETRIC ANALYSIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

FISSION GAS RELEASE + ANALYTICAL TECHNIQUE + FUEL ROD + SAMPLING + KRYPTON + XENON + FOREIGN EXCHANGE

136189
PREUSS HJ
TECHNICAL REPORT CONCERNING PROJECT BMFTRS183. STATUS OF THE INVESTIGATION OF THE ENERGY BALANCE FOLLOWING THE HYPOTHETICAL ROD-FAILURE WHICH ASSUMES RUPTURE OF THE CONCRETE (IN GERMAN)
KRAFTWERK UNION. ERLANGEN. F.R. GERMANY
BMFT RS 183 + GERRSR-218 +. 96 PPS. 6 TABS. 38 FIGS. 29 REFS. MAY 1977

CALCULATIONS WERE MADE TO DESCRIBE THE COURSE OF FOUR MELT DOWN EVENTS AND TO DETERMINE THE DURATION OF THE ACCIDENT. THE ANALYSES FOR THE FOUR ACCIDENT PHASES IN EACH CASE (HEAT UP AT THE CORE. EVAPORATION OF THE RESIDUAL WATER. HEAT UP OF THE RPV. AND PENETRATION OF THE CONCRETE) WERE MADE UNDER THE HYPOTHETICAL ASSUMPTION THAT NO WATER WILL BE INJECTED INTO THE REACTOR CORE DURING THE ENTIRE ACCIDENT. ALSO. PRESSURE AND TEMPERATURE IN THE CONTAINMENT ATMOSPHERE WERE CALCULATED TAKING INTO ACCOUNT THE MASS AND ENERGY TRANSPORT INTO THE CONTAINMENT DURING THE FOUR ACCIDENT PHASES. RESULTS WERE COMPARED WITH WASH-1400 ANALYSES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

GERMANY + *ENERGY + *MATERIAL BALANCE + *ACCIDENT ANALYSIS + ACCIDENT. LOSS OF COOLANT + CONCRETE + CONTAINMENT INTEGRITY + FOREIGN EXCHANGE + ACCIDENT. CORE DISRUPTIVE + CORE MELTDOWN + CONTAINMENT + PRESSURE. INTERNAL

135310
SEIGER H
ULTRASONIC TESTING OF THE SPHERIC APERTURE FIELD REGION OF LIGHT WATER REACTOR PRESSURE VESSELS (IN GERMAN)
KRAUTKRAMER GMBH. F.R. GERMANY
BMFT-RS 2702 + GERRSR-163 +. 90 PPS. FIGS. REFS. APRIL 1975

VECTOR ALGEBRA WAS USED TO CARRY OUT CALCULATIONS CONCERNING THE ULTRASONIC TESTING OF THE SPHERIC APERTURE FIELD REGION OF REACTOR PRESSURE VESSELS AND THE CONSTRUCTION OF TESTING UNITS SUITABLE FOR SUCH TESTS. THE PRACTICAL METHODS WERE DEVELOPED ON THE BASIS OF THE THEORETICAL CONSIDERATION REPORTED IN THIS PAPER AND BY MEASUREMENT IN THE PHILIPPSBOURG II NUCLEAR POWER PLANT. THE APERTURE FIELD REGIONS CONSISTED OF THICK-WALLED SPHERIC SECTIONS PENETRATED BY PIPING AND PROVIDED WITH AUSTENITIC PLATING ON THE INTERNAL SURFACE. THE GENERATION OF CRACKS AT THE PIPING AND AT THE CENTER BETWEEN NEIGHBORING PIPES WAS TESTED BY THE VERTICAL SOUND GENERATION METHOD AND THE PRINCIPLES FOR THE CONSTRUCTION OF TESTING HEADS WERE OUTLINED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

ULTRASONICS + PRESSURE VESSELS + REACTOR. LWR + CRACK + MATHEMATICAL TREATMENT + FOREIGN EXCHANGE

135331
WAGNER + SEIGER
EXPERIMENTAL INVESTIGATION OF THE PROOF OF CRACKS IN PIPING REGION OF REACTOR PRESSURE VESSELS (IN GERMAN)
KRAUTKRAMER GMBH. F.R. GERMANY
BMFT-RS 27/2 + GERRSR-164 +. 20 PPS. 10 FIGS. NOV. 15. 1974

THE FAULT IS INDICATED BY TAKING ADVANTAGE OF THE ANGLE MINOR EFFECT; THAT IS. THE SOUND PULSE HITS THE REFLECTOR IN SUCH A WAY THAT IT IS REFLECTED ON THE INTERNAL SURFACE OF THE VESSEL. COMING BACK IN A LINE THAT IS NEARLY PARALLEL TO THE INCIDENT BEAM. THE TEST SPECIMENS WERE TWICE PLATED; THE NOTCHES ADDED TO THEM SIMULATED CRACKS. THE TEST FAULTS WERE LAID OUT IN SUCH A WAY THAT THEY PRESENTED AN ANGLE OF INCLINATION OF 60 TO 120 DEGREES WITH RESPECT TO THE TESTING HEAD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

CRACK + PIPES AND PIPE FITTINGS + PRESSURE VESSELS + ULTRASONICS + POWER PLANT. NUCLEAR + FAILURE. INSTRUMENT + FOREIGN EXCHANGE

136255
OTTE H-J
DEVELOPMENT OF MANIPULATORS FOR THE REMOTE INSPECTION OF REACTOR PRESSURE VESSELS (IN GERMAN)
M.A.N.. NURNBERG. F.R. GERMANY
RS 27/2 + M.A.N.-NR. 853580 + GERRSR-181 +. 45 PPS. 18 FIGS. 2 REFS. NOV. 1976

DESCRIBES THE WORK CARRIED OUT BY M.A.N. AND SPONSORED BY THE BMFT AS PART OF THE RESEARCH PROJECT ON THE DEVELOPMENT OF MANIPULATORS. MANIPULATORS HAVE BEEN DEVELOPED FOR THE VARIOUS SECTIONS OF THE REACTOR PRESSURE VESSEL. BOTH FOR INTERNAL INSPECTION AND FOR EXTERNAL INSPECTION. IN VIEW OF THE SPACE CONDITIONS IN THE BOTTOM PART OF THE REACTOR PRESSURE VESSEL. INTERNAL INSPECTION IS PREFERRED FOR THE PWR CONCEPT AND AN EXTERNAL INSPECTION FOR THE BWR CONCEPT. MANIPULATORS FOR

136255 *CONTINUED*
EXTERNAL INSPECTION HAVE BEEN DEVELOPED FOR THE TOP CLOSURES OF BOTH CONCEPTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*R AND D PROGRAM + *REMOTE MANIPULATING AND VIEWING + SURVEILLANCE PROGRAM + PRESSURE VESSELS + REACTOR. PWR + REACTOR. BWR + GERMANY + FOREIGN EXCHANGE

136188
HEINRICH D + LINDNER J + MULLER G + REGNET K
RESEARCH ON THE DETECTION OF FLAWS USING REMOTE SUPERSONIC TESTS WITH A LARGE CONTAINER WALL WITH NATURAL OCCURRING FLAWS
M.A.N.. NURNBERG. F.R. GERMANY
RS 27/2 + M.A.N.-NR. 853503 + GERRSR-217 +. 90 PPS. FIGS. SEPT. 1974

NO ENGLISH ABSTRACT WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*FLAW + *SURVEILLANCE PROGRAM + *ACOUSTICS + TESTING + STRUCTURAL INTEGRITY + STRUCTURAL ANALYSIS. DYNAMIC + R AND D PROGRAM + GERMANY + FOREIGN EXCHANGE

132553
AULER I + EDER D + GASCHE A
ANALYSIS OF THE EFFECTS OF SERIOUS ACCIDENTS ON THE DOWNTIME OF NUCLEAR POWER PLANTS. RESEARCH PROJECT RS 155 (IN GERMAN)
NIS. NUKLEAR-INGENIEUR-SERVICE GMBH. FRANKFURT AM MAIN. F.R. GERMANY
BMFT RS 155 + NIS REPORT 203 + GERRSR-167 +. 117 TABS. FIGS. REFS. 1976

TWO HYPOTHETICAL ACCIDENTS CHARACTERIZED BY A DOUBLE-ENDED-RUPTURE OF A MAIN COOLING LOOP ARE CONSIDERED FOR A 1200 MWE PWR REFERENCE PLANT. THE FISSION PRODUCTS. FISSILE MATERIALS AND ACTIVATED MATERIALS RELEASED INTO THE CONTAINMENT UNDER THESE CONDITIONS ARE INVESTIGATED AND ANALYZED ACCORDING TO RADIOLOGICAL CRITERIA. USING SIMPLIFIED DISTRIBUTION MODELS SOME LIMITS ARE ESTIMATED FOR THE CONCENTRATION OF THE ACTIVITY ON THE SURFACES WITHIN THE CONTAINMENT AS WELL AS THE SUMP WATER. THE RESULTING AVERAGE DOSE RATES ARE IN THE ORDER OF MAGNITUDE OF SOME 100 REM/H.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

ACCIDENT ANALYSIS + ACCIDENT. LOSS OF COOLANT + FISSION PRODUCT RELEASE + REACTOR. PWR + GERMANY + RADIOACTIVITY RELEASE + AVAILABILITY + FOREIGN EXCHANGE

138037
MAYINGER F
THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON THERMO- AND FLUIDDYNAMIC BEHAVIOR OF THE REACTOR CORE DURING THE FIRST BLOWDOWN PERIOD (IN GERMAN)
TECHNISCHE UNIVERSITAT HANNOVER. F.R. GERMANY
BMFT-FB 163-01 + GERRSR-237 +. 70 PPS. FIGS. REFS. 1976

ANNUAL REPORT FOR WORK UNDERTAKEN IN 1975. ACHIEVEMENTS INCLUDED COMPLETION OF A TEST-SECTION FOR ENTRAINMENT MEASUREMENTS IN STEADY-STATE ANNULAR FLOW; EARLY INVESTIGATION OF THE ENTRAINMENT BEHAVIOR; THE DEVELOPMENT OF A MEASURING TECHNIQUE TO INVESTIGATE ENTRAINMENT MASS FLOW RATE. DROPLET DISTRIBUTION ACROSS THE CROSS SECTION. AND DROPLET SIZE DISTRIBUTION; AND THE THEORETICAL DEVELOPMENT OF FOUR DIMENSIONLESS NUMBERS WHICH MAY BE VALID IN DESCRIBING THE ENTRAINMENT BEHAVIOR OF HEATED TWO-PHASE FLOW UNDER REACTOR CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*FLOW. TWO PHASE + *BLOWDOWN + *ACCIDENT. LOSS OF COOLANT + COMPARISON. THEORY AND EXPERIENCE + *FLOW THEORY AND EXPERIMENTS + FLOW. ANNULAR + MEASUREMENT + FLOW DISTRIBUTION + INSTRUMENT. FLOW + ANALYTICAL MODEL + *THERMAL HYDRAULIC ANALYSIS + FOREIGN EXCHANGE

132790
VIECENZ H-J + MAYINGER F
PHASE SEPARATION (IN GERMAN)
TECHNISCHEN UNIVERSITAT HANNOVER. F.R. GERMANY
BMFT-FB RS 179-01 + GERRSR-238 +., 62 PPS. FIGS. REFS. 1976

REPORT OF WORK FOR 1975. ACHIEVEMENTS: LIQUID CARRY OVER AT A FREE SURFACE AND THE INTERACTION OF VAPOR AND LIQUID IN TWO-PHASE MIXTURE STEADY STATE VAPOR INJECTION TESTS AND BLOWDOWN EXPERIMENTS OF A VESSEL WERE CONSIDERED FOR THEORETICAL AND EXPERIMENTAL INVESTIGATIONS OF THE PHASE SEPARATION MECHANISM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U.S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

BLOWDOWN + *FLOW. TWO PHASE + *FLOW THEORY AND EXPERIMENTS + COMPUTER PROGRAM + COMPARISON. THEORY AND

132790 *CONTINUED*
EXPERIENCE + VOID FRACTION + DROPLET + FLOW DISTRIBUTION + MEASUREMENT + FOREIGN EXCHANGE

135309
THE "REACTOR PRESSURE VESSEL" RESEARCH PROGRAM. THE 22 NIMOCR 37 HIGH PRIORITY PROGRAM. (IN GERMAN)
UNIVERSITAT STUTTGART, F.R. GERMANY
BMFT-TB RS 101 + GERRSR-158 +. 110 PPS, FIGS, REFS, JULY 1977

THE PROGRAM INVOLVES THE STRENGTH OF WELDMENTS OF FINE-GRAIN STRUCTURAL STEEL 22 NICRMO 37 WHICH HAS BEEN USED FOR REACTOR PRESSURE VESSELS, STEAM GENERATORS AND PIPING SYSTEMS OF GERMAN LIGHT-WATER-REACTOR POWER PLANTS. THE CRITICAL HEAT-AFFECTED AND OVERHEATED AREA WAS SUBJECTED TO LOAD-CAPACITY AND EMBRITTLEMENT TESTS TO OBTAIN DATA FOR SAFETY EVALUATION OF RELAXATION CRACKING. THE DATA IN THE CRITICAL STRUCTURE RANGE MAY REFLECT INCORRECTLY THE TRUE LOCAL RELATIONS. THEREFORE A TECHNIQUE WAS DEVELOPED TO SIMULATE THE STRUCTURAL CHANGES DURING CRACKING; THIS INCLUDED SIMULATION OF OVERHEATING AND ANNEALING. THE WORK INVOLVED TEMPERATURE MEASUREMENTS, PARAMETRIC STUDIES BY SIMULATING THE EVENTS IN THE HEAT ZONE DURING WELDING AND ANNEALING, CHEMICAL ANALYSES, MATERIALS TESTING AND EMBRITTLEMENT STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

PRESSURE VESSELS + STEAM GENERATOR + PIPES AND PIPE FITTINGS + WELDS + STEEL + REACTOR, LWR + SIMULATION + TESTING + FOREIGN EXCHANGE

137390
BECK W + SCHMIDT F + UNGER H
FINAL REPORT ON THE RESEARCH PROGRAM. INTEGRAL ESTIMATE OF THE INFLUENCE OF DATA INACCURACIES OF REACTOR SAFETY CALCULATIONS BY MEANS OF STATISTICAL METHODS (IN GERMAN)
UNIVERSITAT STUTTGART, F.R. GERMANY
BMFT RS 212 + GERRSR-173 +. 135 PPS, TABS, FIGS, APRIL 1977

MONTE CARLO METHODS FOR THE ESTIMATION OF THE INFLUENCE OF STATISTICALLY DISTRIBUTED ERRORS IN THE INPUT DATA OF COMPLEX COMPUTER PROGRAM SYSTEMS ON THE SPREAD OF THE RESULTS OF THE CALCULATIONS HAVE BEEN DEVELOPED IN THE FRAME OF RS 212 (INTAB). WITHIN THE PRESENT PROJECT THE METHODS HAVE BEEN APPLIED FOR BLOWDOWN CALCULATIONS WITH RELAP 3. RESULTS WERE FOR INSTANCE: THE SPREAD OF THE OUTPUT DATA IS OFTEN RELATIVELY HIGH COMPARED TO THAT OF THE INPUT DATA, E.G., BLOWDOWN-TIMES BETWEEN 18 S AND 28 S (AVERAGE 22 S, SPREAD 15%, INPUT DATA SPREAD APPR. 5%).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

MONTE CARLO + SAFETY ANALYSIS + BLOWDOWN + COMPUTER PROGRAM + ERROR ANALYSIS + STATISTICAL ANALYSIS + FOREIGN EXCHANGE + R AND D PROGRAM

136256
RESEARCH PROGRAM ON COMPONENT SAFETY: SUMMARY OF DETAILED SPECIFICAITONS (IN GERMAN)
UNIVERSITAT STUTTGART, F.R. GERMANY
RS 192 + GERRSR-182 +. 115 PPS, 73 FIGS, MAY 1977

NO ENGLISH ABSTRACT WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

POWER PLANT, NUCLEAR + *COMPONENTS + RELIABILITY, COMPONENT + *SAFETY ANALYSIS + TECHNICAL SPECIFICATIONS + *R AND D PROGRAM + FOREIGN EXCHANGE

136285
FOHL J
SUMMARY OF THE SCOPE OF THE HSST-IRRADIATION PROGRAM - PRINCIPALLY REGARDING INVESTIGATIONS OF BRITTLE FRACTURE BEHAVIOR (IN GERMAN)
UNIVERSITAT STUTTGART, F.R. GERMANY
BMFT 313-5691-RS-186 + GERRSR-186 +. APPROX. 85 PPS, TABS, FIGS, MAY 1977

NO ENGLISH ABSTRACT PROVIDED AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + R AND D PROGRAM + STEEL + *IRRADIATION TESTING + *BRITTLE FRACTURE + PRESSURE VESSELS + METALLURGY + FOREIGN EXCHANGE

136286
FOHL J
SUMMARY OF THE HSST PROGRAM REGARDING RESEARCH ON VESSEL WELDS (IN GERMAN)
UNIVERSITAT STUTTGART, F.R. GERMANY
BMFT 313-5691-RS 186 + GERRSR-187 +. APPROX. 61 PPS, 12 TABS, 35 FIGS, 12 REFS, MAY 1977

136286 *CONTINUED*
NO ENGLISH ABSTRACT PROVIDED AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + R AND D PROGRAM + STEEL + *PRESSURE VESSELS + *WELDS + METALLURGY + FOREIGN EXCHANGE

3. JAPANESE LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM JAPAN DURING THE FIRST HALF OF 1978 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

134238
STUDY ON FLAW SIZE MEASUREMENT BY ACOUSTICAL HOLOGRAPHY (IN JAPANESE)
MITSUBISHI HEAVY INDUSTRIES CO. LTD., JAPAN
REPORT 52-STAE-194 + JPNRSR-99 +. 34 PPS. FIGS. REFS. 1976

IN THIS STUDY, THE ACOUSTICAL HOLOGRAPHY TECHNIQUE WAS EVALUATED FOR THE SIZING OF FATIGUE CRACKS AND ARTIFICIAL FLAWS INTRODUCED IN THICK WALL SECTION AND SAFE-END WELDS IN NUCLEAR REACTOR PRESSURE VESSELS. THE RESULTS ARE SUMMARIZED AS FOLLOWS: (1) THE SIZE OF FATIGUE CRACKS INTRODUCED IN 150 MM THICK STEEL PLATE COULD BE MEASURED BY ACOUSTICAL HOLOGRAPHY WITHIN A VARIABILITY OF +/- 3 MM. (2) THE ARTIFICIAL FLAWS INTRODUCED IN AUSTENITIC STEEL SAFE-END WELDS, COULD BE DETECTED BY ULTRASONIC EXAMINATION, BUT ACOUSTICAL HOLOGRAPHY SHOWED POOR ABILITY IN SIZING OF FLAWS DUE TO THE EFFECTS OF THE MICROSTRUCTURE OF AUSTENITIC WELDS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STEEL, STAINLESS + ULTRASONICS + WELDS + PLATE + FLAW + HOLOGRAPHY + STEEL + FOREIGN EXCHANGE

134237
STUDY ON THE BEHAVIOR OF THE LOCAL STRUCTURE OF NUCLEAR REACTOR PIPING FOR ITS SAFETY EVALUATION (IN JAPANESE)
JAPAN WELDING ENGINEERING SOCIETY
JWES-AE-77-6 + JPNRSR-100 +. 191 PPS. TABS. FIGS. 1976

IN TESTS RUN ON PIPING, NO CRACK PROPAGATION WAS OBSERVED EVEN WHEN THE PIPE WAS LOADED TO THE YIELD POINT. A THERMAL SHOCK EXPERIMENT SHOWS THAT CRACKS OCCUR NOT ON THE SURFACE, BUT AT A POINT JUST BELOW THE SURFACE. IN WELDING TYPE 304 PIPES, IT WAS FOUND THAT RESIDUAL STRESSES AT THE INNER SURFACE ARE GREAT ENOUGH TO INDUCE STRESS CORROSION CRACKING, AND THAT THE HIGHER WELDING HEAT INPUT CAUSES HIGHER SENSITIZATION. TESTS SHOW THAT SENSITIZED STAINLESS STEELS SUCH AS TYPES 321 AND 347 ARE HIGHLY RESISTANT TO STRESS CORROSION CRACKING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STEEL, STAINLESS + STRESS CORROSION + CRACK + PIPES AND PIPE FITTINGS + WELDS + FOREIGN EXCHANGE

134236
STUDY ON THE SAFETY ASSESSMENT OF FATIGUE STRENGTH OF PRESSURE VESSELS AND PIPING SYSTEMS IN NUCLEAR PLANTS (IN JAPANESE)
JAPAN WELDING ENGINEERING SOCIETY
JWES-AE-7707 + JPNRSR-101 +. 277 PPS. TABS. FIGS. 1976

THE STUDY WAS CARRIED OUT TO EXAMINE THE FATIGUE CRACK GROWTH FROM MULTIPLE ADJACENT FLAWS, WHICH ARE OFTEN DETECTED IN REAL VESSELS AND PIPINGS, AND TO CONTRIBUTE TO THE ESTABLISHMENT OF A RATIONAL PROCEDURE FOR THE SAFETY EVALUATION OF NUCLEAR PLANTS. WIDE PLATE SPECIMENS 20 MM THICK AND APPROXIMATELY 200 MM WIDE WITH SINGLE OR MULTIPLE SURFACE NOTCHES WERE PREPARED FROM A A533B STEEL PLATE. AS THE FIRST STEP OF STUDY, SIXTEEN COMBINATIONS OF NOTCH ARRANGEMENTS WERE CHOSEN AS TEST NOTCH CONDITIONS. FATIGUE TESTS WERE PERFORMED UNDER AXIAL TENSION, SIMPLE BENDING AND BIAXIAL TENSION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FATIGUE + PRESSURE VESSELS + POWER PLANT, NUCLEAR + SAFETY EVALUATION + CRACK + GROWTH/DEVELOPMENT + FLAW + FOREIGN EXCHANGE

135409
SUZUKI M + MATSUMOTO I + ADACHI H + SUZUKI M
ROSA-II TEST DATA REPORT, 8 (RUNS 324, 325, 326), EFFECTS OF DOWNCOMER GAP AND ECCS INJECTION FLOW RATE (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7236 + JPNRSR-102 +. 137 PPS. TABS. FIGS. SEPT. 1977

RESULTS OF THE ROSA-II TEST SIMULATING A LOSS-OF-COOLANT ACCIDENT (LOCA) AND EFFECT OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) IN A PRESSURIZED WATER REACTOR (PWR) ARE PRESENTED INCLUDING THE TEST CONDITIONS AND INTERPRETATIONS OF THE PHENOMENA FOR TEST RUNS 324, 325 AND 326. EACH TEST WAS MADE IN A DOUBLE-ENDED GUILLOTINE BREAK OF THE COLD LEG. EFFECTS OF THE FOLLOWING PARAMETERS ON THE COOLANT BEHAVIOR IN BOTH BLOWDOWN AND REFLOODING PHASES WERE EXAMINED IN COMPARING BETWEEN RUNS, PREVIOUS TEST WITH THE SAME ECCS INJECTION POSITION, ETC: 1) INCREASE OF THE DOWNCOMER GAP, 2) INCREASE OF ECCS INJECTION FLOW RATE, 3) ENLARGEMENT OF THE OUTLET FLOW AREA OF PUMP IN THE BLOWDOWN LOOP. INCREASING THE DOWNCOMER GAP, THE EFFECT OF COUNTER CURRENT FLOW LIMIT (CCFL) DECREASED AND WATER ACCUMULATION INCREASED. BY INCREASING THE ECC WATER INJECTION RATE BY 1.5 TIMES THE NORMAL, THE RATE OF WATER ACCUMULATION INCREASED BY 6 TIMES THE NORMAL. NOTE: TEXT IN JAPANESE, FIGURE LEGENDS IN ENGLISH.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + *ACCIDENT, LOSS OF COOLANT + *EMERGENCY COOLING SYSTEM + *BLOWDOWN + SIMULATION + CORE REFLOODING + FOREIGN EXCHANGE

135408
SOBAJIMA M + ADACHI M + SUZUKI M + MURATA H
ROSA-II TEST DATA REPORT, 9 - THE LARGEST HOT LEG BREAKS - (RUNS 418, 419, 420, AND 423) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7239 + JPNRSR-103 +. 177 PPS. TABS. FIGS. SEPT. 1977

RESULTS OF THE ROSA-II TEST SIMULATING A LOSS-OF-COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR (PWR) ARE PRESENTED, INCLUDING THE TEST CONDITIONS AND INTERPRETATION OF THE PHENOMENA FOR TEST RUNS 418, 419, 420 AND 423. IN THE CASE OF A HOT LEG BREAK, CORE WATER IS LOST A FEW SECONDS AFTER THE BREAK AND TEMPERATURE OF THE FUEL RODS RISES DUE TO EXPOSURE. HOWEVER, WATER ACCUMULATION EFFECT IN THE PRESSURE VESSEL WITH ACCUMULATORS IS RELATIVELY LARGE, SINCE THE COUNTER-CURRENT FLOW IN DOWNCOMER IS SMALL; REFLOODING IS THUS ATTAINED EARLY. WHEN POWER SUPPLY TO THE CORE IS CONTINUED, QUENCHING OF THE RODS IS DELAYED AND THE WATER LEVEL LOWERED. COOLING OF THE UPPER AND MIDDLE CORE IS MAINTAINED WITH GRADUAL RECOVERY OF THE WATER LEVEL DUE TO LPCI INJECTION. NOTE: TEXT IN JAPANESE, FIGURE LEGENDS IN ENGLISH.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + *ACCIDENT, LOSS OF COOLANT + *EMERGENCY COOLING SYSTEM + SIMULATION + CORE REFLOODING + *BLOWDOWN + FOREIGN EXCHANGE

135845
MARAYAMA Y + IZUMI F
FREG-3: A COMPUTER PROGRAM TO PREDICT THE FUEL-TO-CLADDING HEAT TRANSFER COEFFICIENT IN ACCORDANCE WITH FUEL ROD IRRADIATION HISTORY - CALCULATION PROCEDURES AND MODELS (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7278 + JPNRSR-104 +. 60 PPS. 12 TABS. 9 FIGS. 30 REFS. SEPT. 1977

FREG-3 PREDICTS THE TEMPERATURE DISTRIBUTION IN A FUEL ROD AND THE STORED ENERGY BASED ON THE DISTRIBUTION. THE TEMPERATURE DISTRIBUTION IS CALCULATED IN ACCORDANCE WITH FUEL-ROD IRRADIATION HISTORY UNDER NORMAL OPERATING CONDITIONS. AN IMPORTANT FACTOR IN OBTAINING THE TEMPERATURE DISTRIBUTION IS THE GAP HEAT TRANSFER COEFFICIENT BETWEEN PELLET OUTER SURFACE AND CLADDING INNER SURFACE. TO OBTAIN THE PROPER COEFFICIENT, THE PROGRAM INCORPORATES SUBMODELS AND CORRELATIONS. PURPOSE OF THE PROGRAM IS TO EVALUATE STORED ENERGY IN THE FUEL ROD PRECEDING THE INITIATION OF THE LOCA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + HEAT TRANSFER ANALYSIS + HEAT CONDUCTANCE, FUEL TO CLAD + FUEL ROD + *HEAT TRANSFER COEFFICIENT + CLADDING + DISTRIBUTION + TEMPERATURE + FOREIGN EXCHANGE

135846
SODA K + SASAKI S + AKIMOTO M
PREDICTION OF LOFT LI-4 EXPERIMENT - CSNI STANDARD PROBLEM NO. 5 (IN ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7329 + JPNRSR-106 +. 97 PPS. 6 TABS. 58 FIGS. 16 REFS. OCT. 1977

LOFT LI-4 EXPERIMENTAL RESULTS WERE PREDICTED BY LOFT ANALYSIS GROUP AND CODE DEVELOPMENT GROUP USING RELAP-4J AND ALARM-P1 RESPECTIVELY. THE INPUT DATA PREPARED BY THE FORMER GROUP WERE USED IN BOTH THE ANALYSES. THUS ANY DIFFERENCES IN THE RESULTS SHOULD STEM FROM THE DIFFERENCES IN CODE PERFORMANCE CHARACTERISTICS OF THE TWO CODES. SOME RESULTS ARE: (1) THE COOLANT BEHAVIORS PREDICTED BY RELAP-4J AND ALARM-P1 ARE IN GOOD AGREEMENT ALTHOUGH SOME DIFFERENCES DO EXIST BETWEEN THESE TWO CALCULATION MODELS. (2) LARGE DIFFERENCE IS SEEN IN COOLANT FLOW RATE ACROSS THE PUMP. THE COAST DOWN AND THE FLOW RATE BY ALARM-P1 ARE LARGER AND SMALLER RESPECTIVELY THAN BY RELAP-4J.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF FLOW + PUMPS + COMPUTER PROGRAM + PERFORMANCE + HYDRAULIC EXPERIMENT + FLOW THEORY AND EXPERIMENTS + COMPARISON + FOREIGN EXCHANGE

135847
TAKEDA T
USERS GUIDE FOR CODAC-NO. 5: A COMPUTER PROGRAM TO CALCULATE NUCLIDE YIELDS IN COMPLEX DECAY CHAIN FOR SELECTION OF OPTIMUM IRRADIATION AND COOLING CONDITION (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7392 + JPNRSR-107 +. 184 PPS. FIGS. NOV. 1977

IN THIS CODE, EIGHTEEN KINDS OF VALUABLE TABLES AND GRAPHS CAN BE PREPARED FOR OUTPUT. THEY ARE AVAILABLE FOR SELECTION OF OPTIMUM IRRADIATION AND COOLING CONDITIONS AND FOR OTHER INTENTIONS IN ACCORDANCE WITH IRRADIATION AND COOLING. FOR AN EXAMPLE, THE RATIO OF A NUCLIDE YIELD TO TOTAL NUCLIDE YIELD DEPENDING ON IRRADIATION AND COOLING TIMES IS OBTAINED. IN THESE OUTPUTS, SEVERAL KINDS OF COMPLEX AND INTRICATE EQUATIONS ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

135047 *CONTINUED*
COMPUTER PROGRAM + RADIONUCLIDE + RADIOISOTOPE + OPTIMIZATION + ACTIVATION + IRRADIATION TESTING + FOREIGN EXCHANGE

135411
KUROYANAGI. T + IWAMURA T
TRANSIENT BEHAVIOR OF PRESSURE AND WALL TEMPERATURE IN WATER FLOW IN A TUBE UNDER FLOW REDUCTION CONDITIONS (TRANSIENT FLOW BOILING EXPERIMENTS AT AN ATMOSPHERIC PRESSURE (III)) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST.. TOKYO
JAERI-M-7396 + JPNRSR-108 +. 61 PPS. TABS. FIGS. NOV. 1977

FLOW REDUCTION TRANSIENT BOILING PHENOMENA HAVE BEEN STUDIED IN VERTICAL UP-FLOW OF WATER IN A TUBE UNDER AN ATMOSPHERIC PRESSURE TO PROVIDE INFORMATION ON THE TRANSIENT THERMAL-HYDRAULIC BEHAVIOR OF A LWR CORE DURING POWER COOLING MISMATCH (PCM) CONDITIONS. THE ELECTRICALLY HEATED TEST SECTION WAS STAINLESS STEEL TUBE 8 MM IN DIAMETER AND 800 MM LONG. ACCORDING TO THE OBSERVED BEHAVIOR OF THE PRESSURE AND WALL TEMPERATURE. THE FLOW REDUCTION TRANSIENT BOILING PHENOMENA ARE SIMILAR TO THOSE IN THE STEADY-STATE BOILING AT FLOW REDUCTION RATES BELOW ABOUT 10CM/SEC/SEC UNDER THE RANGES OF EXPERIMENTAL CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

FLOW. TWO PHASE + FLOW THEORY AND EXPERIMENTS + BOILING + *TEMPERATURE + THERMAL TRANSIENT + FOREIGN EXCHANGE + *PRESSURE TRANSIENT

135848
OHKI N + HARAYAMA Y + TAKEDA T + IZUMI F
EVALUATION OF FISSION-PRODUCT GASES IN PROGRAM GAPCON SERIES AND FREG-3 TO ESTIMATE THE GAP HEAT TRANSFER COEFFICIENT (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST.. TOKAI
JAERI-M-7417 + JPNRSR-109 +. 30 PPS. 6 TABS. 12 FIGS. 13 REFS. DEC. 1977

MAKES A COMPARISON OF THE AMOUNT OF XENON AND KRYPTON AS CALCULATED BY THE GAPCON SERIES AND FREG-3 COMPUTER PROGRAMS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

COMPUTER PROGRAM + HEAT TRANSFER ANALYSIS + HEAT CONDUCTANCE. FUEL TO CLAD + FISSION PRODUCT. VOLATILE + HEAT TRANSFER COEFFICIENT + COMPARISON + THERMAL CONDUCTIVITY + FOREIGN EXCHANGE

136240
SOBAJIMA M + ADACHI M + SUZUKI M + MATSUMOTO I
ROSA-II TEST DATA REPORT. 10 - EFFECTS OF SECONDARY SYSTEM AND HPCI SYSTEM IN SMALL BREAKS (RUNS 415. 417. 421. 422) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST.. TOKAI
JAERI-M-7437 + JPNRSR-110 +. 151 PPS. FIGS. REFS. DEC. 1977

RESULTS OF THE ROSA-II TEST SIMULATING A LOSS-OF-COOLANT ACCIDENT (LOCA) IN A LIGHT WATER REACTOR (LWR) ARE PRESENTED FOR TEST RUNS 415. 417. 421 AND 422. EVEN IN SMALL BREAK AT THE COLD LEG. THE CORE IS EXPOSED TO VOID AND THE TEMPERATURE RISES. IN A SMALL BREAK OF THE HOT LEG. HOWEVER. CORE COOLING KEEPS WITHOUT TEMPERATURE RISE. BECAUSE THERE STILL REMAINS MUCH RESIDUAL WATER AND UPWARD CORE FLOW EXISTS. DIRECT EFFECT OF THE HPCI ON THE DEPRESSURIZATION RATE IS SMALL. BUT IT INCREASES THE ACCUMULATOR INJECTION RATE. LEADING TO EARLY CORE REFLOODING AND EARLY CORE COOLING FROM UPWARD. EFFECTS OF THE SECONDARY SYSTEM DEPRESSURIZATION ARE INCREASE OF DEPRESSURIZATION AND DISCHARGE RATES OF THE PRIMARY LOOP. WHICH RESULTS IN EARLY INITIATION OF THE ACCUMULATOR INJECTION AND CORE REFLOODING. (NOTE: ALTHOUGH TEXT IS IN JAPANESE. FIGURES AND TABLES IN ENGLISH.)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

*ACCIDENT. LOSS OF COOLANT + REACTOR. LWR + CORE REFLOODING + *HPCI + FLOW THEORY AND EXPERIMENTS + HEAT TRANSFER EXPERIMENT + *COOLING SYSTEM. SECONDARY + *EMERGENCY COOLING SYSTEM + FOREIGN EXCHANGE

137615
TANAKA M
A SURVEY OF THE RESEARCH WORKS ON LWR CONTAINMENT SPRAY EFFECTS (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST.. TOKAI
JAERI-M-7481 + JPNRSR-111 +. 151 PPS. TABS. FIGS. 49 REFS. JAN. 1978

THIS IS A SUMMARY OF CONTAINMENT SPRAY RESEARCH RESULTS FROM JAPAN. THE U.S.. AND EUROPE. THE TEXT IS IN JAPANESE. FIGURES AND REFERENCES ARE IN ENGLISH. THE TEXT APPARENTLY CONSISTS OF EXTRACTS OR SUMMARIES OF THE REPORTS LISTED AT THE END OF THE REPORT. THE REPORT SUMMARIZES MOST OF THE RESEARCH THAT HAS BEEN DONE THROUGHOUT THE WORLD IN THE PAST; IT IS NOT CLEAR IF THE RESULTS OF THE VARIOUS PROGRAMS ARE EVALUATED OR COMPARED. OR ASSESSED RELATIVE TO ITS APPLICABILITY IN THE JAPANESE LWR PROGRAM.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

137615 *CONTINUED*
REACTOR, LWR + REACTOR, SAFETY RESEARCH + ACCIDENT, LOSS OF COOLANT + CONTAINMENT SPRAY + IODINE + TEMPERATURE + PARTICLE + FOREIGN EXCHANGE + FISSION PRODUCT, IODINE + CONTAINMENT, PRESSURE RELIEF

137774
TANAKA M + MIMORI T
GAS-WATER PARTITION OF IODINE IN BWR PLANTS AND JAERI MODEL CONTAINMENT VESSEL (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7482 + JPNRSR-112 +. 50 PPS. FIGS. 34 REFS. JAN. 1978

REMOVAL OF RADIOIODINE FROM BWR PLANTS AND JAERI MODEL CONTAINMENT VESSEL BY WATER SPRAYING WAS CALCULATED WITH THE COMPUTER PROGRAM SPRINKLE PREPARED BY ONE OF THE AUTHORS FOR CARRYING OUT CONTAINMENT SPRAY EXPERIMENTS. THE FOLLOWING WERE STUDIED QUANTITATIVELY: 1) EFFECT OF CONTAINMENT SPRAY, 2) EQUILIBRIUM STATE DOMINATION IN IODINE LEAKAGE FROM THE CONTAINMENT VESSEL, 3) THE TIME ELAPSING FROM CONTAINMENT SPRAY START TO EQUILIBRIUM STATE, AND 4) EQUILIBRIUM PARTITION COEFFICIENT IN THE CONTAINMENT VESSEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTAINMENT + CONTAINMENT SPRAY + IODINE + *PARTITION COEFFICIENT + REACTOR, BWR + WASHOUT + REACTOR, LWR + ACCIDENT, LOSS OF COOLANT + ACCIDENT, HYPOTHETICAL + FOREIGN EXCHANGE

137775
TANAKA M
PRELIMINARY CALCULATIONS WITH CODE CONTEMPT-LT FOR SPRAY COOLING TESTS WITH JAERI MODEL CONTAINMENT VESSEL (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7483 + JPNRSR-113 +. 38 PPS. 5 TABS. 31 FIGS. 6 REFS. JAN. 1978

LWR PLANTS HAVE A CONTAINMENT SPRAY SYSTEM TO REDUCE THE ESCAPE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT IN A LOSS-OF-COOLANT ACCIDENT (LOCA) BY WASHING OUT FISSION PRODUCTS, ESPECIALLY RADIOIODINE, AND CONDENSING THE STEAM TO LOWER THE PRESSURE. FOR CARRYING OUT THE CONTAINMENT SPRAY TESTS, PRESSURE AND TEMPERATURE BEHAVIOR OF THE JAERI MODEL CONTAINMENT VESSEL IN SPRAY COOLING HAS BEEN CALCULATED WITH COMPUTER PROGRAM CONTEMPT-LT. THE FOLLOWING WERE STUDIED QUANTITATIVELY: (1) PRESSURE AND TEMPERATURE RAISE RATES FOR STEAM ADDITION RATE AND (2) PRESSURE FALL RATE FOR SPRAY FLOW RATE AND SPRAY HEAT TRANSFER EFFICIENCY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + REACTOR, LWR + CONTAINMENT + CONTAINMENT SPRAY + PRESSURE TRANSIENT + THERMAL TRANSIENT + FOREIGN EXCHANGE

137776
KITAGUCHI H + SUZUKI M + SOBAJIMA M
PRELIMINARY ANALYSIS OF ROSA III EXPERIMENT (II) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7488 + JPNRSR-114 +. 63 PPS. 7 TABS. 118 FIGS. FEB. 1978

LOSS-OF-COOLANT ACCIDENT (LOCA) EXPERIMENTS TO BE PERFORMED IN ROSA III HAVE BEEN EXAMINED WITH COMPUTER CODE RELAP-4J CONCERNING THE EXPERIMENTAL CONDITIONS. THE RESULTS (1) TO (3), AND THE NEEDS (4) TO (6) ARE PRESENTED. (1) INITIAL ENTHALPY DISTRIBUTION IS IMPORTANT FOR SIMULATION OF BREAK FLOW OF AN ACTUAL BWR. (2) THE SIMULATIONS OF LOWER PLENUM FLASHING AND PRESSURE TRANSIENT IN PRESSURE VESSEL ARE GOOD EXCEPT WHEN POWER IS LACKING. (3) THE SIMULATION OF THE CLADDING TEMPERATURE TRANSIENT IS DIFFICULT BECAUSE OF LACK OF PHYSICAL PROPERTIES. (4) THE INITIAL PRESSURE DISTRIBUTION IN THE FACILITY FOR DIFFERENT CORE FLOW RATES UP TO 72 LB/SEC MUST BE ATTAINED TO ANALYZE ACCURATELY. (5) REVERSE CORE FLOW DETECTORS AND REVERSE JET PUMP FLOW DETECTORS ARE NECESSARY TO COMPARE FLOW PATTERN OF RECIRCULATION LOOPS BETWEEN CALCULATION AND EXPERIMENT. (6) FURTHER INFORMATION IS NECESSARY ON PHYSICAL PROPERTIES OF THE FUELS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + REACTOR, BWR + ACCIDENT, LOSS OF COOLANT + CLADDING + TEMPERATURE + PRESSURE TRANSIENT + SIMULATION + FLOW, RECIRCULATION + PUMP, JET + FOREIGN EXCHANGE

136197
KUROYANAGI T + IWAMURA T
BURNOUT PHENOMENA IN A TUBE DURING FLOW REDUCTION TRANSIENTS (TRANSIENT FLOW BOILING UNDER ATMOSPHERIC PRESSURE (IV)) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7489 + JPNRSR-115 +. 84 PPS. TABS. FIGS. FEB. 1978

BURNOUT PHENOMENA IN FLOW REDUCTION TRANSIENTS HAVE BEEN STUDIED IN UP-FLOW OF WATER IN A VERTICAL TUBE UNDER ATMOSPHERIC PRESSURE. THE ELECTRICALLY HEATED TEST SECTION IS A STAINLESS STEEL TUBE 8 MM IN DIAMETER AND 800 MM LONG. EXPERIMENTAL CONDITIONS ARE: INLET WATER TEMPERATURE = 30 TO 80 DEG C; HEAT FLUX = 0.87 TO 1.16 X 10(EXP 6) KCAL/HM SQUARED; INITIAL MASS VELOCITY = 1.8 TO 3.8 X 10(EXP 6) KG/HM SQUARED; FLOW REDUCTION TIME = 0.07 TO 70 SEC; AND FLOW REDUCTION RATE =

136197 *CONTINUED*

0.24 TO 1100 CM/SEC/SEC. BEYOND A FLOW REDUCTION RATE OF 10 CM/SEC/SEC, THERE APPEAR PHENOMENA PECULIAR TO THE FLOW REDUCTION TRANSIENT BURNOUT IN CHANNEL PRESSURE AND WALL TEMPERATURE. THE BURNOUT INLET MASS VELOCITY IS ALSO LOWER THAN THE STEADY-STATE ONE UNDER THE SAME HEATED POWER CONDITIONS. ALTHOUGH TEXT IS IN JAPANESE, FIGURES, AND TABLES ARE IN ENGLISH.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FLOW, TWO PHASE + *HEAT FLUX, BURNOUT + *HEAT FLUX, CRITICAL + *FLOW THEORY AND EXPERIMENTS + HEAT TRANSFER EXPERIMENT + BOILING + TRANSIENT + COMPARISON, THEORY AND EXPERIENCE + ANALYTICAL MODEL + FOREIGN EXCHANGE

137466

PRELIMINARY ANALYSIS OF DOWNCOMER EFFECTIVE WATER HEAD DURING REFLOOD PHASE IN PWR LOCA (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7490 + JPNRSR-116 +. 103 PPS. 63 FIGS. 26 REFS. FEB. 1978

THE RESULTS ARE DESCRIBED OF PRELIMINARY ANALYSIS ON THE DOWNCOMER EFFECTIVE WATER HEAD, WHICH IS THE DRIVING FORCE TO FEED THE EMERGENCY COOLANT INTO THE CORE DURING THE REFLOOD PHASE IN PWR LOCA. DUE TO THE VAPOR GENERATED BY HEAT RELEASE FROM THE DOWNCOMER WALLS (REACTOR VESSEL ETC.), A TWO-PHASE FLOW APPEARS IN THE DOWNCOMER, SO THAT THE DOWNCOMER EFFECTIVE WATER HEAD IS REDUCED. TAKING INTO CONSIDERATION THERMO-HYDRAULIC CHARACTERISTICS OF THE DOWNCOMER, THE SENSITIVITIES WERE STUDIED CONCERNING FACTORS INFLUENCING THE EFFECTIVE WATER HEAD, AND THE CONCEPTUAL HISTORIES WERE OBTAINED OF THE DOWNCOMER EFFECTIVE WATER HEAD DURING THE REFLOOD PHASE. GEOMETRICAL SCALINGS, RUN PARAMETERS AND TEST PROCEDURES WERE ALSO OBTAINED ON THE BASIS OF THE RESULTS OF PRELIMINARY ANALYSIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FOREIGN EXCHANGE

137348

KUKITA Y
ANALYSIS OF THE SHORT-TERM PRESSURE RESPONSE OF MARK II CONTAINMENT TO A LOSS-OF-COOLANT ACCIDENT (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7504 + JPNRSR-117 +. 31 PPS. FIGS. 13 REFS. JAN. 1978

PRESSURE RESPONSE OF THE MARK II PRESSURE SUPPRESSION CONTAINMENT DURING THE VERY EARLY STAGE OF A LOSS-OF-COOLANT ACCIDENT WAS STUDIED TO OBTAIN A FIRM BASIS FOR EVALUATION OF THE PRESSURE SUPPRESSION POOL SWELL. CALCULATIONS WERE MADE BY CONTEMPT-LT/022 AND AN EXTENDED VERSION OF CONTEMPT-LT WHICH CALCULATES THE CONTAINMENT PRESSURE RESPONSE SIMULTANEOUSLY WITH THE POOL SWELL. IT HAS BEEN PROVED THAT TAKING ACCOUNT OF THE INITIAL INVENTORY OF THE BROKEN LINE AND/OR POOL SWELL LEADS TO MORE CONSERVATIVE EVALUATION OF THE DRYWELL PRESSURE DURING THE POOL SWELL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BWR + CONTAINMENT, PRESSURE SUPPRESSION + RESPONSE TIME + CONTAINMENT, HIGH PRESSURE + ACCIDENT, LOSS OF COOLANT + COMPUTER PROGRAM + FOREIGN EXCHANGE

137777

SUZUKI M + MATSUMOTO I + ADACHI M
ROSA-II TEST DATA REPORT, II -- EFFECTS OF BREAK AREA DISTRIBUTION AND CIRCULATION PUMP ON CORE FLOW - (RUNS 327, 328, 329, 330) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7505 + JPNRSR-118 +. 162 PPS. TABS. FIGS. REFS. FEB. 1978

RESULTS OF THE ROSA-II TESTS SIMULATING A LOSS-OF-COOLANT ACCIDENT (LOCA) AND EFFECTS OF AN EMERGENCY CORE COOLING SYSTEM (ECCS) IN A PRESSURIZED WATER REACTOR (PWR) ARE PRESENTED INCLUDING THE TEST CONDITIONS AND INTERPRETATIONS OF THE DATA IN TEST RUNS 327, 328, 329 AND 330. EACH TEST WAS PERFORMED WITH LARGE DOUBLE-ENDED HOT LEG BREAK AND EFFECT OF THE BREAK AREA DISTRIBUTION (BREAK DIAMETER ARE 28.0 MM AT ONE END AND 37.5 MM AT THE OTHER END OF BREAK) AND OF PUMP CIRCULATION UPON COOLANT FLOW IN THE CORE WERE STUDIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + CORE REFLOODING + BLOWDOWN + FOREIGN EXCHANGE + EMERGENCY COOLING SYSTEM

137747

MOTIZUKI Y + SOBAJIMA M + SUZUKI M
LOCA ANALYSIS PROGRAM RELAP4J FOR WATER COOLED NUCLEAR REACTORS (MODIFICATION OF RELAP4-MOD2) (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7506 + JPNRSR-119 +. 77 PPS. TABS. FIGS. FEB. 1978

RELAP4J, A VERSION OF RELAP4-MOD2 BASED ON ROSA-I AND II TESTS HAS THE FOLLOWING ALTERATIONS:

137747 *CONTINUED*

DISCHARGE COEFFICIENT CORRELATION IN DISCHARGE FLOW CALCULATION. ZALOWDEK'S EQUATION FOR SUBCOOLED DISCHARGE FLOW. SONIC CHOKE INSTEAD OF MOODY CHOKE IN HIGH QUALITY REGIONS. WILSON'S EQUATION FOR BUBBLE RISING VELOCITY. BUBBLE RISE MODEL FOR COUNTER-CURRENT FLOW. STEADY-STATE PRESSURE DISTRIBUTION CALCULATION. STABILIZED NUMERICAL INTEGRATION AND UNIT CONVERSION FROM FT-LB TO MKS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

COMPUTER PROGRAM + BUBBLE + REACTOR. LWR + CORRELATION + FLOW BLOCKAGE + FLOW + EMERGENCY COOLING SYSTEM + FOREIGN EXCHANGE

137469
OHNISHI N + TANZAWA S + TANZAWA T + KITANO T
ENERGY DEPOSITION IN NSRR TEST FUELS (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH ESTABLISHMENT
JAERI-M-7539 + JPNRSR-120 +. 70 PPS. TABS. FIGS. FEB. 1978

INTERPRETATION OF FUEL PERFORMANCE DATA COLLECTED DURING INPILE TESTING IN THE NSRR REQUIRES A KNOWLEDGE OF THE ENERGY DEPOSITION OR ENTHALPY INCREASE IN EACH SAMPLE TESTED. THE REPORT DESCRIBES THE RESULTS OF ABSOLUTE MEASUREMENT OF FISSION PRODUCTS AND CONTENTS OF URANIUM IN IRRADIATED TEST FUELS WHICH WERE PERFORMED TO DETERMINE THE ENERGY DEPOSITION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

ENERGY + IN PILE EXPERIMENT + DATA COLLECTION + THERMODYNAMICS + IRRADIATION TESTING + FOREIGN EXCHANGE + DEPOSITION

137768
NISHIO G + TAMURA T + TANAKA M
LWR CONTAINMENT SPRAY IODINE REMOVAL CALCULATION CODE--MIRA-PB (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST.. TOKAI
JAERI-M-7594 + JPNRSR-121 +. 113 PPS. FIGS. 43 REFS. MARCH 1978

THE COMPUTER CODE MIRA-PB FOR PREDICTING IODINE REMOVAL BY CONTAINMENT SPRAY IS MADE ON THE BASIS OF MIRAP/MIRAB CODE DEVELOPED IN BATTELLE'S COLUMBUS LABORATORIES. MIRA-PB CONSIDERS BEHAVIOR OF ELEMENTAL IODINE. ORGANIC IODIDE. AND IODIC AEROSOL IN NATURAL DEPOSITION. LIQUID-FILM ABSORPTION. SPRAY WASHOUT. FILTRATION AND LEAKAGE TO ATMOSPHERE. IODINE REMOVAL BY SPRAYS IN PWR. BWR. BATTELLE'S CSE AND JAERI-MODEL CONTAINMENT IS CALCULATED BY MIRA-PB UNDER LOCA SIMULATED CONDITIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

REACTOR. LWR + ACCIDENT. LOSS OF COOLANT + PARTITION COEFFICIENT + AIR CLEANING + CONTAINMENT + IODINE + ORGANIC IODIDE + CONTAINMENT SPRAY + SPRAY + WASHOUT + FOREIGN EXCHANGE + COMPUTER PROGRAM

137778
NISHIO G + TAMURA T + TANAKA M
COMPUTER PROGRAM OF IODINE REMOVAL IN THE LWR CONTAINMENT VESSEL UNDER LOCA CONDITIONS. MIRA-PB (IN ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST.. TOKAI
JAERI-M-7602 + JPNRSR-122 +. 41 PPS. MARCH 1978

LWR PLANTS HAVE A CONTAINMENT SYSTEM FOR REACTOR SAFETY CONSISTING OF SPRAY AND AIR CLEANING FILTER. R.L. RITZMAN OF BATTELLE COLUMBUS LAB. DEVELOPED COMPUTER CODE MIRAP/MIRAB FOR PREDICTING IODINE REMOVAL BY CONTAINMENT SYSTEM FOR PWR AND BWR. THE COMPUTER CODE MIRA-PB, A MODIFICATION OF MIRAP/MIRAB, IS DESCRIBED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

COMPUTER PROGRAM + CONTAINMENT + IODINE + CONTAINMENT SPRAY + AIR CLEANING + REACTOR. LWR + FOREIGN EXCHANGE

4. JAPANESE FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM JAPAN DURING THE FIRST HALF OF 1978 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

138635
KIKUCHI Y + HAGA K
TRANSIENT BOILING OF SODIUM IN A SINGLE-PIN GEOMETRY UNDER LOSS-OF-FLOW CONDITIONS (IN ENGLISH)
POWER REACTOR & NUCLEAR FUEL DEVELOPMENT CORP., JAPAN
PNC-N241 78-09 + JPNRSR-142F +. 11 PPS, 10 FIGS, 15 REFS, FEB. 1978

TRANSIENT SODIUM BOILING EXPERIMENTS HAVE BEEN CONDUCTED IN AN ELECTRICALLY HEATED SINGLE-PIN ANNULAR CHANNEL UNDER LOSS-OF-FLOW CONDITIONS. EACH RUN WAS MADE BY REDUCING OR STOPPING THE FLOW AT CONSTANT HEATER POWER. RECORDS WERE OBTAINED OF THE CHANGES OF TEMPERATURES, FLOW RATES, PRESSURES AND VOIDS. THE EXPERIMENTAL RESULTS WERE COMPARED WITH ANALYTICAL CALCULATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*SODIUM + *ACCIDENT, LOSS OF FLOW + TRANSIENT + REACTOR, LMFBR + *BOILING + HEAT TRANSFER EXPERIMENT + VOID + VOID FRACTION + PHASE CHANGE + FOREIGN EXCHANGE

137468
NISHIO G + KITANI S + TAKADA J
SODIUM OXIDE AEROSOL BEHAVIOR IN A CLOSED CHAMBER UNDER THERMAL CONVECTION FLOW (IN ENGLISH)
POWER REACTOR & NUCLEAR FUEL DEVELOPMENT CORP., TOKAI
PNC J250 75-14 TR + JPNRSR-144F +. 27 PPS, 2 TABS, 11 FIGS, MARCH 1978

THE PRESENT WORK WAS UNDERTAKEN TO OBTAIN THE EFFECT OF THERMOPHORETIC ADHERENCE OF NUCLEAR AEROSOL ON THE REACTOR CONTAINMENTS IN THE EVENT OF A HYPOTHETICAL ACCIDENT OF AN LMFBR. IN THE ACCIDENT, SODIUM OXIDE AEROSOL CONTAINING PUO2-UO2 FUEL AND FISSION PRODUCTS MAY BE SUSPENDED IN THE CONTAINMENT AND TENDS TO DEPOSIT ON THE COLD WALL OF THE CONTAINMENT, SO THAT THE AEROSOL CONCENTRATION DECREASES BY THERMOPHORESIS IN ADDITION OF SETTLING AND DIFFUSIONAL PLATING. EXPERIMENTS WERE MADE TO DETERMINE THE RELATION FOR THE DEGREE OF THERMOPHORETIC ADHERENCE OF SODIUM OXIDE AEROSOL IN A CHAMBER. THE RESULTS OBTAINED BY THE EXPERIMENT WERE COMPARED WITH THOSE OF THE CALCULATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AEROSOL + DEPOSITION + THERMOPHORESIS + DIFFUSION + REACTOR, LMFBR + SODIUM + OXIDE + PLUTONIUM + FOREIGN EXCHANGE + COMPARISON, THEORY AND EXPERIENCE

5. UK LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE UK DURING THE FIRST HALF OF 1978 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

136902
BOGARTZ BD + GOODMAN AME + O'BRIEN TP + RAWLINGSON M + EDWARDS AR
DEPRESSURIZATION STUDIES PHASE 2: RESULTS OF TESTS 115 AND 130
ATOMIC WEAPONS RESEARCH ESTABLISHMENT, ALDERMASTON, UK
AWRE/44/88/147 + SRD-R-115 +. 105 PPS, 3 TABS, 82 FIGS, JAN. 1978

A BASIC EXPERIMENTAL PROGRAMME INVOLVING THE SUDDEN DEPRESSURIZATION OF A SIMPLE PIPE SYSTEM CONTAINING WATER AT PRESSURES AND TEMPERATURES IN THE RANGE 3.45 TO 17.24 MPA (500 TO 2500 PSIG) AND 220 TO 250 DEG C RESPECTIVELY, HAS RECENTLY BEEN CONCLUDED. THIS REPORT IS THE FIRST IN A SERIES TO GIVE DETAILS OF THE RESULTS OF TESTS. THIS SERIES WAS DESIGNED TO EXAMINE THE EFFECT OF ADDING A RESERVOIR TO THE PIPE USED IN THE PHASE I SERIES AND THEREBY TO PROVIDE ADDITIONAL INFORMATION FOR DEVELOPMENT OF A CALCULATIONAL PROGRAM BEING DEVELOPED BY SRD.

AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

JACOBS + *ACCIDENT, LOSS OF PRESSURE + *ACCIDENT, LOSS OF COOLANT + MEASUREMENT + FLOW + FLOW, TWO PHASE + R AND D PROGRAM + EQUIPMENT DEVELOPMENT + INSTRUMENT, FLOW + INSTRUMENT, PRESSURE + INSTRUMENT, TEMPERATURE

137779
BRYCE WM + GREEN C
THE DEFINITION AND EVALUATION OF THE REWETTING PROCEDURE IN THE LOCA CODE RELAP-UK MK IV
UKAEA ATOMIC ENERGY ESTABLISHMENT, WINFRITH, UK
AEEW-R-1156 + UKRSR-11 +. 43 PPS, 3 TABS, 12 FIGS, 8 REFS, FEB. 1978

THIS REPORT DESCRIBES THE PRESENT STATE OF THE REWETTING PROCEDURE WHICH HAS BEEN DEVELOPED FOR THE GENERAL LOSS-OF-COOLANT ACCIDENT CODE RELAP-UK MK IV. AN EXPRESSION FOR THE VELOCITY OF A WETTING FRONT ON A DRY FUEL PIN IS GIVEN TOGETHER WITH A SET OF CONDITIONS UNDER WHICH THE FRONT CAN EXIST AND PROGRESS. THE REWETTING VELOCITY EXPRESSION USED IS BASED ON A PUBLISHED CONDUCTION-CONTROLLED REWETTING FORMALISM. THE RELAP-UK MODEL DEVELOPED IS USED TO SIMULATE A FEW RELEVANT EXPERIMENTS. THE COMPARISONS SHOW THAT THE REWETTING PROCEDURE TENDS TO BE PESSIMISTIC BUT THAT6 IT IS A SIGNIFICANT IMPROVEMENT IN THE MODELLING ABILITY OF RELAP-UK. AS A RESULT OF THE SIMULATION STUDIES POSSIBLE FUTURE DEVELOPMENTS ARE SUGGESTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM + *WETTING + FUEL ROD + SIMULATION + COMPARISON, THEORY AND EXPERIENCE + HEAT FLUX, DRYOUT + HEAT TRANSFER EXPERIMENT + THERMAL EXPERIMENT + FOREIGN EXCHANGE

138830
RALPH JC + ROBERTS JC
FREE CONVECTION HEAT TRANSFER MEASUREMENTS IN HORIZONTAL LIQUID LAYERS WITH INTERNAL HEAT GENERATION
UKAEA ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, UK
AERE-R-7841 + UKRSR-20 +. 24 PPS, 11 FIGS, 16 REFS, OCT. 1974

HEAT TRANSFER AND DETAILED TEMPERATURE MEASUREMENTS HAVE BEEN MADE ON A RECTANGULAR TANK OF WEAK ELECTROLYTE HEATED ELECTRICALLY THROUGHOUT ITS VOLUME AND COOLED THROUGH THE TOP BOUNDING SURFACE. A CORRELATION BETWEEN A SUITABLY DEFINED NUSSELT NUMBER AND THE RAYLEIGH NUMBER WAS ESTABLISHED WHICH SHOWED THAT THE NUSSELT NUMBER INCREASED WITH RAYLEIGH NUMBER TO THE QUARTER POWER. AS A CONSEQUENCE OF THIS THE BOUNDARY LAYER TEMPERATURE DROP IS PROPORTIONAL TO THE COOLING SURFACE HEAT FLUX ONLY, TO THE 0.75 POWER. COMPARISON WITH SIMILAR WORK CARRIED OUT ELSEWHERE SHOWED THAT ANY NET HEAT TRANSPORT DOWNWARDS THROUGH A SIMULTANEOUSLY COOLED LOWER SURFACE IS RELATIVELY SMALL AND HAS LITTLE EFFECT ON THE BULK TEMPERATURE OF THE TURBULENT FLUID.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*HEAT TRANSFER, NATURAL CONVECTION + *HEAT GENERATION, INTERNAL + HEAT TRANSFER EXPERIMENT + *MEASUREMENT + HEAT TRANSFER COEFFICIENT + CORRELATION + FOREIGN EXCHANGE

137746
MATHER DJ
HUBBLE-BUBBLE I: A COMPUTER PROGRAM FOR THE ANALYSIS OF NON-EQUILIBRIUM FLOWS OF WATER
UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, UK
SRD-R-97 + UKRSR-29 +. 27 PPS, 4 FIGS, 3 REFS, FEB. 1978

THIS REPORT DESCRIBES THE COMPUTER PROGRAM HUBBLE-BUBBLE I WHICH SIMULATES THE NON-EQUILIBRIUM FLOW OF WATER AND STEAM IN A PIPE. THE CODE IS DESIGNED TO EXAMINE THE TRANSIENT FLOW DEVELOPING IN A PIPE CONTAINING HOT COMPRESSED WATER FOLLOWING THE RUPTURE OF A RETAINING DIAPHRAGM. ALLOWANCE IS MADE FOR AN AREA CHANGE IN THE PIPE. PARTICULAR ATTENTION IS PAID TO THE NON-EQUILIBRIUM DEVELOPMENT OF VAPOUR BUBBLES AND TO THE TRANSITION FROM A BUBBLE-LIQUID REGIME TO A DROPLET-VAPOUR REGIME. THE MATHEMATICAL AND COMPUTATIONAL MODEL IS DESCRIBED TOGETHER WITH A SUMMARY OF THE FORTRAN SUBROUTINES AND LISTING OF DATA INPUT.

AVAILABILITY - THE EDITOR, UNITED KINGDOM ATOMIC ENERGY AUTHORITY, SAFETY & RELIABILITY DIRECTORATE, CULCHETH, WARRINGTON WA3 4NE, ENGLAND

COMPUTER PROGRAM + BUBBLE + DROPLET + STEAM + FLOW + FLOW, TWO PHASE + TRANSIENT + PRESSURE TRANSIENT +

137746 *CONTINUED*
FOREIGN EXCHANGE

138033
PECKOVER RS + TURLAND BD + WHIPPLE RTP
ON THE GROWTH OF MELTING POOLS IN SACRIFICIAL MATERIALS
UKAEA CULHAM LAB., OXON, UK
CLM-P-511 + UKRSR-31 +. 8 PPS, 6 TABS, 5 FIGS, 10 REFS, 1977

THE GROWTH OF THE MELT POOL IN NATURAL BEDROCK OR CHOSEN SACRIFICIAL MATERIAL IS CONSIDERED FOR THE HYPOTHETICAL CASE IN WHICH HEAT PRODUCING CORE DEBRIS PENETRATES THE BOTTOM OF THE REACTOR VESSEL FOLLOWING A HYPOTHETICAL CORE MELTDOWN IN A FISSION REACTOR. THE CASE ANALYZED ASSUMES THAT THE MOLTEN POOL AND MATERIAL ARE MISCIBLE. THE COMPUTER CODES BASALT AND PAMPUR ARE USED IN THE ANALYSIS. SEVERAL CANDIDATE SACRIFICIAL BED MATERIALS HAVE BEEN CONSIDERED FOR INSERTION IN A CAVITY; THE MOST PROMISING CONFIGURATION CONSISTS OF A LOW MELTING POINT ROCK SURROUNDED BY A MORE REFRACTORY INSULATOR, PROVIDED THAT COOLING OF THE UPPER SURFACE OF THE POOL CAN BE RELIED UPON.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*CORE MELTDOWN + *ACCIDENT, HYPOTHETICAL + COMPUTER PROGRAM + *ANALYTICAL MODEL + ACCIDENT ANALYSIS + ACCIDENT, CONSEQUENCES + ACCIDENT, FUEL SLUMP + CONTAINMENT ANALYSIS + CONTAINMENT DESIGN + PHASE CHANGE + REACTOR, LWR + REACTOR, LMFBR + FOREIGN EXCHANGE

138034
TURLAND BD + PECKOVER RS + DULLFORCE TA
ON THE STABILITY OF ADVANCING MELTING FRONTS TO SPATIAL PERTURBATIONS
UKAEA CULHAM LAB., OXON, UK
CLM-P-512 K+ UKRSR-32 +. 7 PPS, 1 TAB, 7 FIGS, 4 REFS, NOV. 1977

THE STABILITY OF A PLANE INTERFACE BETWEEN A SOLID AND A LIQUID PHASE IS CONSIDERED FOR THE CASE WHEN AN EQUILIBRIUM STEADY UNIFORM HEAT FLUX DENSITY IS APPLIED NORMAL TO THE INTERFACE. THE ANALYSIS SHOWS THAT THE INTERFACE IS STABLE AGAINST SMALL SPATIAL PERTURBATIONS WHEN THE INTERFACE MOVES AT A STEADY UNIFORM VELOCITY MU AND THE LIQUID PHASE CLOSE TO THE INTERFACE EITHER (A) DOES NOT CONVECT OR (B) BEHAVES AS AN INCOMPRESSIBLE INVISCID MEDIUM. THE RESULTS HOLD WHEN THE APPLIED UNIFORM HEAT FLUX DENSITY KIS CHANGING WITH TIME, PROVIDED THAT THE TIMESCALE ASSOCIATED WITH THIS VARIATION IS LONG COMPARED WITH THE TIME IN WHICH SPATIAL PERTURBATIONS DECAY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CORE MELTDOWN + HEAT TRANSFER EXPERIMENT + COMPARISON, THEORY AND EXPERIENCE + ANALYTICAL MODEL + STABILITY + PHASE CHANGE + ACCIDENT, HYPOTHETICAL + ACCIDENT ANALYSIS + ACCIDENT, CONSEQUENCES + ACCIDENT, FUEL SLUMP + CONTAINMENT ANALYSIS + CONTAINMENT DESIGN + REACTOR, LWR + REACTOR, LMFBR + FOREIGN EXCHANGE

137969
PECKOVER RS
DIRECTIONAL DEPENDENCE AND NON-UNIFORMITY OF JOULE HEATING IN NATURAL CONVECTION EXPERIMENTS
UKAEA CULHAM LAB., OXON, UK
CLM-RR/S2/15 + UKRSR-35 +. 12 PPS, 2 FIGS, 14 REFS, NO DATE

VOLUMETRIC HEATING OF A HORIZONTAL LAYER OF FLUID TO PRODUCE NATURAL CONVECTION IS USUALLY CARRIED OUT USING JOULE HEATING. IF THE CURRENT FLOWS HORIZONTALLY, THE HEAT SOURCE DENSITY IS DEFICIENT IN BOTH THE UPPER AND LOWER BOUNDARY LAYERS. IF THE CURRENT FLOWS VERTICALLY THE HEAT SOURCE DENSITY IS ENHANCED IN THESE BOUNDARY LAYERS. AT HIGH RAYLEIGH NUMBERS THAT EFFECT CAN BE IGNORED IN THE THIN UPPER BOUNDARY LAYER. IN THE LOWER BOUNDARY LAYER WHICH IS THICKER AND HAS A MUCH SMALLER NUSSELT NUMBER, THESE DEPARTURES OF THE HEAT SOURCE DENSITY FROM UNIFORMITY CAN SIGNIFICANTLY DISTORT THE DEPENDENCE OF NUSSELT NUMBER ON THE RAYLEIGH NUMBER. FIRST ORDER CORRECTION FORMULAE ARE PRESENTED AND APPLIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FLOW + BOUNDARY LAYER + HEAT TRANSFER, NATURAL CONVECTION + HYDRAULIC EXPERIMENT + THERMAL EXPERIMENT + FOREIGN EXCHANGE

6. UK FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE UK DURING THE FIRST HALF OF 1978 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

136893
WILLIAMS RJ
REVIEW OF AEROSOL PROBLEMS AND THE THEORY OF AEROSOL PHYSICS WITH PARTICULAR REFERENCE TO SODIUM COOLED FAST REACTORS
UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, UK
SRD-R-82 +. 37 PPS, 4 TABS, 29 REFS, JAN. 1978

PROCESSES THAT WOULD GOVERN THE DEVELOPMENT, TRANSPORT, AND REMOVAL OF AEROSOLS, WHICH ARE OF INTEREST IN THE STUDY OF HYPOTHETICAL CORE DISRUPTIVE SITUATIONS IN POOL TYPE SODIUM COOLED FAST REACTORS, ARE DISCUSSED. THEORETICAL DESCRIPTIONS OF THESE PROCESSES ARE PRESENTED AND KNOWN INADEQUACIES INDICATED. THE INTERPRETATION OF EXPERIMENTAL DATA AND NUMERIC SOLUTION OF THE GOVERNING EQUATIONS IS BRIEFLY CONSIDERED.

AVAILABILITY - THE EDITOR, JNITED KINGDOM ATOMIC ENERGY AUTHORITY, SAFETY & RELIABILITY DIRECTORATE, CULCHETH, WARRINGTON WA3 4NE, ENGLAND

AEROSOL + REACTOR, LMFBR + PROPERTY, PHYSICAL + SODIUM + TRANSPORT PROPERTY + AGGLOMERATE + UNITED KINGDOM + JACOBS

138028
DULLFORCE TA + REYNOLDS JA + PECKOVER RS
INTERFACE TEMPERATURE CRITERIA AND THE SPONTANEOUS TRIGGERING OF SMALL-SCALE FUEL-COOLANT INTERACTIONS
UKAEA CULHAM LAB., OXON, UK
CLM-P-517 + UKRSR-9 +. 5 PPS, 2 FIGS, 5 REFS, JAN. 1978

SEVERAL CRITERIA HAVE BEEN SUGGESTED FOR THE ONSET OF FUEL-COOLANT INTERACTIONS THE MOST IMPORTANT BEING BASED ON THE INTERFACE TEMPERATURE, T(SUB I), HAVING TO EXCEED SOME FIXED VALUE SUCH AS THE FUEL MELTING POINT. THESE ARE BRIEFLY DISCUSSED AND IT IS SHOWN THAT IN THREE SYSTEMS WITH LOW MELTING POINT FUELS (I.E. LOW COMPARED TO THE COOLANT HOMOGENEOUS NUCLEATION TEMPERATURE, T(SUB HOM) THE T(SUB I) = T(SUB HOM) CRITERION MUST BE EXCEEDED FOR SPONTANEOUS FCIS TO OCCUR IN SMALL DROP EXPERIMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*FUEL COOLANT INTERACTION + INTERFACE + BUBBLE NUCLEATION + REACTOR, LMFBR + TEMPERATURE + *HEAT TRANSFER EXPERIMENT + FOREIGN EXCHANGE

138029
VAUGHAN GJ + BRISCOE F
THE DEFINITION OF EFFICIENCY FOR FUEL COOLANT INTERACTIONS
UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, UK
SRD R 99 + UKRSR-17 +. 12 PPS, 12 REFS, FEB. 1978

IN THE ANALYSIS OF FUEL COOLANT INTERACTIONS, THE EFFICIENCY OF THE CONVERSION OF THE AVAILABLE ENERGY IN THE INITIAL CONFIGURATION INTO MECHANICAL WORK IS OFTEN QUOTED. HOWEVER, THE METHOD OF CALCULATION OF THIS QUANTITY EMPLOYED BY VARIOUS AUTHORS IS DIFFERENT GIVING RISE TO POSSIBLE CONFUSION WHEN COMPARING DIFFERENT EXPERIMENTS. VARIOUS METHODS OF CALCULATION ARE REVIEWED AND A STANDARD DEFINITION OF EFFICIENCY IS PROPOSED WHICH CAN BE USED IN ALL SITUATIONS, BASED ON THE MAXIMUM THEORETICAL WORK CALCULATED FROM THE HICKS-MENZIES MODEL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*FUEL COOLANT INTERACTION + REACTOR, LMFBR + *ANALYTICAL MODEL + ANALYTICAL TECHNIQUE + FOREIGN EXCHANGE

137600
SMITH RD
A REVIEW OF THE UK FAST REACTOR PROGRAMME: MARCH 1978
UKAEA, RISLEY ESTABLISHMENT, UK
NOR-3686 + UKRSR-18 +. 47 PPS, MARCH 1978

THE FOLLOWING TOPICS ARE INCLUDED IN THE REVIEW: THE DOUNREAY FAST REACTOR (DFR); THE PROTOTYPE FAST REACTOR (PFR); THE COMMERCIAL DEMONSTRATION FAST REACTOR (CDFR); ENGINEERING DEVELOPMENT; MATERIALS DEVELOPMENT; CHEMICAL ENGINEERING/SODIUM DEVELOPMENT; FAST REACTOR FUEL; FUEL CYCLE; SAFETY; AND REACTOR PERFORMANCE STUDIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

UNITED KINGDOM + *REACTOR, FAST + *R AND D PROGRAM + REACTOR, LMFBR + *REVIEW + SODIUM + TECHNOLOGY + FUEL, NUCLEAR + SAFETY ANALYSIS + REACTOR DYNAMICS + REACTOR PHYSICS + FOREIGN EXCHANGE

138031
TURLAND BD
THE USE OF A HIGHLY CONDUCTING LINER IN REDUCING LOCAL HEAT FLUX DENSITY CONCENTRATIONS
UKAEA CULHAM LAB., OXON, UK
UKRSR-22F +. 5 PPS, 3 TABS, 2 FIGS, 4 REFS (NO DATE)

138031 *CONTINUED*

THE SUGGESTION THAT A CORE-CATCHER FOR USE IN A FAST REACTOR MAY BE PROTECTED BY A HIGHLY CONDUCTING LINER FROM LOCAL FAILURE CAUSED BY NONUNIFORM ACCUMULATION OF HEAT-PRODUCING DEBRIS IS INVESTIGATED. THE EFFECT OF THE LINER IS TO REDUCE THE AMPLITUDE OF MODULATIONS OF THE HEAT FLUX DENSITY AT THE UPPER SURFACE OF THE CORE-CATCHER. A TRANSMISSION COEFFICIENT FOR HEAT FLUX DENSITY ACROSS THE LINER IS DERIVED FOR FOURIER MODES AND THIS IS APPLIED TO DEBRIS ACCUMULATIONS IN ONE AND TWO DIMENSIONS. THE PROTECTION BY A LAYER OF COPPER OF A STAINLESS STEEL CATCHER COOLED FROM SINGLE LINER WILL ONLY SIGNIFICANTLY REDUCE HEAT FLUX DENSITY CONCENTRATIONS WITH TYPICAL WIDTHS LESS THAN TEN TIMES THE LINER THICKNESS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*CORE MELTDOWN ♦ *CORE CATCHER ♦ REACTOR, LMFBR ♦ *ANALYTICAL MODEL ♦ STEEL LINER ♦ HEAT GENERATION, NONUNIFORM ♦ ACCIDENT, HYPOTHETICAL ♦ ACCIDENT ANALYSIS ♦ ACCIDENT, FUEL SLUMP ♦ CONTAINMENT ANALYSIS ♦ CONTAINMENT DESIGN ♦ FOREIGN EXCHANGE

138032
PECKOVER RS ♦ RALPH JC ♦ MCGREEVY R
EXPERIMENTS IN TURBULENT THERMAL CONVECTION DRIVEN BY INTERNAL HEAT SOURCES
UKAEA CULHAM LAB., OXON, UK
FREC(CD)/P(77)26 ♦ UKRSR-24F ♦. 13 PPS, 10 FIGS, 18 REFS, 1977

ANALOGUE LABORATORY EXPERIMENTS HAVE BEEN PERFORMED USING TANKS FILLED WITH WEAK CHLORIDE SOLUTIONS. THE INTERNAL HEATING IN THE PRIMARY EXPERIMENTS HAS BEEN VOLUMETRIC JOULE HEATING WITH HEAT EXTRACTION TAKING PLACE THROUGH THE TEMPERATURE CONTROLLED HORIZONTAL BOUNDING SURFACES. OPERATING CONDITIONS IN THE TESTS EXTENDED WELL INTO THE TURBULENT REGIME AND TEMPERATURES WITHIN THE CONVECTING FLUID WERE FOUND TO BE STRONGLY TIME DEPENDENT. MEAN TEMPERATURE PROFILES OBTAINED USING TRAVERSING THERMOCOUPLES AND FLOW VISUALIZATION FILMS INDICATED A RELATIVELY THICK BOUNDARY LAYER ABOVE THE BOTTOM COOLING SURFACE WITH A THINNER LAYER BELOW THE TOP COOLING SURFACE FROM WHICH PLUMES INTERMITTENTLY BREAK AWAY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LMFBR ♦ HEAT TRANSFER EXPERIMENT ♦ COMPARISON, THEORY AND EXPERIENCE ♦ HEAT TRANSFER, CONVECTION ♦ *TURBULENCE, CONVECTIVE ♦ FOREIGN EXCHANGE ♦ *HEAT GENERATION, INTERNAL

137795
ADLAM JH
COMPUTER CALCULATION OF NATURAL CONVECTION WITH UNIFORM HEAT GENERATION IN THE LIQUID
UKAEA CULHAM LAB., OXON, UK
CLM-M-103 ♦ UKRSR-27F ♦. 37 PPS, 10 FIGS, SEPT. 1975

CALCULATIONS HAVE BEEN MADE WHICH CAN BE COMPARED WITH THE PREVIOUS EXPERIMENTAL RESULTS OF RALPH AND ROBERTS AND OF WILKIE AND FISHER. THE SIMPLIFYING ASSUMPTIONS OF TWO-DIMENSIONAL FLOW HAS BEEN MADE FOR THE CALCULATION. THERE IS NO SIGNIFICANT DIFFERENCE BETWEEN THIS CALCULATION AND THE PREVIOUS EXPERIMENTAL RESULTS CONSIDERED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM ♦ HEAT TRANSFER, NATURAL CONVECTION ♦ COMPARISON, THEORY AND EXPERIENCE ♦ HEAT TRANSFER ANALYSIS ♦ FOREIGN EXCHANGE

137796
TURLAND BD ♦ PECKOVER RS
THE DISTRIBUTON OF FISSICN PRODUCT DECAY HEAT
UKAEA CULHAM LAB., OXON, UK
CLM-P-510 ♦ UKRSR-28F ♦. 10 PPS, PRESENTED AT 3RD POST ACCIDENT HEAT REMOVAL CONFERENCE; ARGONNE, ILL., NOV. 3, 1977

DATA FOR THE DECAY HEAT PRODUCED BY EACH FISSICN PRODUCT ELEMENT IN A FAST REACTOR ARE GIVEN FOR VARIOUS TIMES BETWEEN 20 MINUTES AND 4 MONTHS AFTER SHUTDOWN OF THE CHAIN REACTION. THESE NEW DATA, OBTAINED FROM THE FISPIN CODE, ARE COMPARED WITH THE PREVIOUS ESTIMATES AND THEIR RELEVANCE IN PAHR STUDIES EXAMINED. FOLLOWING A CORE-MELTDOWN IT IS ESTIMATED THAT AT TIMES LESS THAN A FEW HOURS 20% OF THE DECAY HEATING IS BY VOLATILES, WHILE UP TO 30% MAY OCCUR IN A MOLTEN PHASE. AFTER TEN DAYS THE CONTRIBUTION OF VOLATILES BECOME NEGLIGIBLE, BUT 20% OF THE HEAT INPUT MAY STILL BE IN THE STEEL DEBRIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*DECAY HEAT ♦ COMPUTER PROGRAM ♦ REACTOR, FAST ♦ REACTOR, LWR ♦ FOREIGN EXCHANGE

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